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# Evaluation of Uncertainties in Irradiated Hardware Characterization

Prepared by N. Bedore, A. Levin, P. Tuite

Waste Management Group, Inc.

Prepared for
U.S. Nuclear Regulatory
Commission

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# ABSTRACT

Under contract to the U. S. Nuclear Regulatory Commission, Waste Management Group, Inc. has evaluated the techniques used by industry to characterize and classify irradiated hardware components for disposal in accordance with the requirements of 10CFR Part 61 and the uncertainties associated with application of these techniques. This report describes the current practices used to characterize the radionuclide content of hardware components, identifies the uncertainties associated with the techniques and practices considered, and recommends areas for improvement which could reduce uncertainty.

This study indicates that industry uses two different characterization methods. The first uses a combination of gamma scanning, direct sampling, underwater radiation profiling and radiochemical analysis to determine radionuclide content, while the second uses a form of activation analysis in conjunction with underwater radiation profiling. Both methods employ two distinct steps; (1) the determination of Cobalt 60 content, and (2) the determination of scaling factors for hard-to-detect Part 61 radionuclides. There are uncertainties associated with each step for both methods. The accurate determination of Cobalt-60 is critical since the Part 61 activation product radionuclides which affect Part 61 classification are scaled from Cobalt-60. Current uncertainties in Cobalt-60 determination can be reduced by improving underwater radiation profiling equipment and techniques. The calculational techniques used for activation analysis can also be refined to reduce the uncertainties with Cobalt-60 determination. The uncertainties associated with scaling factor determination generally lead to overestimates of some Part 61 radionuclides. Improved radiochemical analysis techniques, and the development of a statistically significant set of component composition data would eliminate some of the conservatism in current scaling factor determination methods.

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# EXECUTIVE SUMMARY

The nuclear industry presently uses several different methods to characterize the radionuclide content of hardware irradiated during normal reactor operations, and to subsequently classify this hardware for disposal under the NRC regulations found in Title 10, Part 61, of the Code of Federal Regulations. Section 61.55 of these regulations includes two tables (Table 1 for long-lived nuclides, and Table 2 for short-lived nuclides) listing limiting concentrations for a number of radionuclides. The study performed herein discusses current practices and methodologies, identifies uncertainties associated with them, and recommends areas for improvements.

The classification of components irradiated in a reactor core, under 10CFR Part 61 requirements, is driven equally by the $^{63}$Ni concentration relative to limits in Table 2, and the combined concentrations of $^{14}$C, $^{59}$Ni, $^{94}$Nb and $^{99}$Tc relative to limits in Table 1. Iron-55, which appears in significant quantities in irradiated hardware, has no practical effect on classification. It does, however, influence disposal costs by impacting the total curie content of the irradiated component.

Hardware irradiated in nuclear reactor cores is categorized into two groups. The first group consists of non-fuel assembly hardware. This group includes source assemblies, reactor control cluster assemblies, thimble plugs, in-core instrumentation guide tubes, and burnable poison rod assemblies in pressurized water reactors; and control rod blades, flow channels, poison plates, and instrument guide tubes in boiling water reactors. The second group consists of fuel disassembly hardware, including assembly end fittings, grid spacers, guide tubes and miscellaneous spacers, and springs and screws for both types of light water reactors.

The concentrations of radionuclides affecting hardware classification depend upon the component's composition and core residence time. A major constituent in most PWR and BWR components, excluding BWR flow channels, is 300 series stainless steel. The stainless steel is sometimes combined with other materials such as boron carbide, borosilicate glass or silver-indium-cadmium alloys to perform a special function within the core. Some PWR and BWR components also contain relatively large amounts of 600 and 700 series inconel. Neutron irradiation of both stainless steel and inconel produces radioactive nickel isotopes with the higher nickel content of the inconels a significant contributor to the activated nickel production. Generally, most stainless steel components with typical core residence times can satisfy Part 61 Class C limits for disposal, particularly when packaged with other types of components. BWR flow channels and fuel assembly guide tubes are composed of Zircaloy 2 and 4, and the activation of these materials is well below that for components which are dominated by stainless steel and/or inconel. Zircaloy components can be readily disposed of in accordance with Part 61 limits. Inconel components can, in general, exceed Class C limits with minimal in-core residence.

The methods used to characterize irradiated components can be divided into two categories which are referred to as the direct sample method and the activation analysis method. Different techniques are applied within each method. Both methods apply a two step approach which relies on first determining the $^{60}$Co content of the component and then determining the relationship between $^{60}$Co and the other hard-to-detect radionuclides of interest. They also rely on underwater profiles of individual components to determine the $^{60}$Co values from which other radionuclides of interest are scaled. The results of this study indicate that improvements can be made in both the analytical and field applications of both characterization methods.

The determination of the $^{60}$Co content is the first critical step in the characterization process. For the direct sample method this initially involves a combination of gamma ray spectroscopy, component sampling and analysis and contact dose profiling of preselected representative components. Once the selected components are characterized, the calculated $^{60}$Co dose rate-to-curie conversion factor is used with individual component radiation profiles to characterize, at a later date, all components of the same type and vintage. Uncertainties are primarily due to potential field practice inconsistencies between initial efforts on representative components and later efforts which characterize the bulk of the component inventory. Uncertainties can also arise from the use of contact dose rate measurements, which raises questions of measurement representativeness given the presence of hot and cold spots on some components, and from instrument use.

There are also $^{60}$Co uncertainties associated with the activation analysis techniques. These techniques minimize the hot spot concern since contact radiation profiles are not used. They do, however, rely on code calculated $^{60}$Co dose rate-to-curie conversion factors which are themselves a source of uncertainty not found in the direct sample method. Potential practice inconsistencies arising from performing the direct sample characterization process in two stages are not present. The use of non-contact radiation profiles also increases the potential error due to source/detector geometry variations relative to the direct sample method.

For both methods some of the sources of uncertainty can be reduced by performing studies which vary instruments and source/detector distances and geometries for the same component types under controlled field conditions. The empirical data obtained from these investigations could quantify the sensitivity of dose rate measurements at the contact and non-contact distances used by both methods for different instruments and instrument configurations. It could also quantify the relative importance of the potential hot spot variances that could result from the simulated point source geometry used in the direct sample method to obtain the $^{60}$Co dose-to-curie conversion factor. Some work on the point kernel shielding calculations could also reduce the uncertainty in the calculational methods used to determine $^{60}$Co using the activation analysis techniques.

Scaling factor determination for both methods also involves uncertainties. The direct sample method determines the scaling factors for most Part 61 radionuclides of significance by analyzing the component sample which is obtained after the gamma spectroscopy is initially performed. The uncertainties in the analysis of these samples is that normally incident to the laboratory methods used to perform the radionuclide specific analysis. There is also an uncertainty associated with the representativeness of the sample that is analyzed since relatively small samples, 10-100 mg, are considered typical for relatively large amounts of hardware. Some radionuclides are scaled from these sample data and others are at such low concentrations that they are below the limit of detection and LLD values are reported.

One of the results of this study indicated that the scaling factors used to estimate the Table 1 radionuclide $^{59}$Ni from the $^{63}$Ni result are consistently at least a factor of 2.5 higher than the scaling factors for $^{59}$Ni calculated using activation analysis techniques for the same type of component. The study also showed that another important Table 1 radionuclide, $^{94}$Nb, is often estimated using LLD values. Both of these scaling factor related uncertainties can lead to a conservative determination of the sum of the fractions for Table 1 radionuclides. Further investigation of both the $^{59}$Ni/$^{63}$Ni relationship and the use of LLD's for $^{94}$Nb and other radionuclides is needed.

The activation analysis method and the different techniques which apply it also have uncertainties in scaling factor estimates. First, this method relies on the relative content of elemental cobalt to the content of other elemental materials such as nickel, niobium, nitrogen, and molybdenum to estimate scaling factors for Part 61 radionuclides. Except for nickel, all of these metals are usually present in trace quantities in most of the materials used for hardware. Thus the uncertainties arise from the unavailability of materials composition data for each batch of components and the variability of the data that is available. The effect of this uncertainty is somewhat mitigated by the method's use of relative compositions as opposed to absolute compositions. Second, the method relies on having information that depicts exposure of the component(s) to neutron flux or irradiation history data as well as the fluxes seen by the component. Core history data is generally available by component and is usually available for batches of components. Available exposure data is maintained for fuel management purposes but there is uncertainty when these data are applied to hardware components. There is also uncertainty associated with the neutron fluxes obtained from fuel management data. The effect of this flux uncertainty is mitigated by the method's use of relative fluxes.

The results of this study show that the Part 61 radionuclide scaling factors obtained using this method are reasonably compatible with those obtained using the direct sample method, except for $^{59}$Ni as noted above, and $^{94}$Nb. The study indicates that the activation analysis techniques tend to overpredict $^{94}$Nb quantities.

The largest uncertainties are those associated with materials composition, particularly the cobalt and niobium. The impact of the cobalt uncertainty is important because it affects the calculation of all scaling factors. The lower the cobalt compositions the higher the scaling factors and the more conservative the results. The niobium is generated in relatively low concentrations but is important because it also has a relatively low Part 61 concentration limit. This coupled with the use of LLD's using the direct sample method make niobium a radionuclide of particular concern. Further investigations of materials composition data could lead to a statistically significant data base from which both the materials composition and LLD uncertainties could be reduced.

## 1.0 INTRODUCTION

Irradiated hardware generated at commercial light water reactors (LWR's) represents a unique form of low-level radioactive waste. Hardware volumes are relatively small but the radiation levels and concentrations of short-lived radionuclides are relatively large. Hardware discharged from the reactor core is normally placed in the facility's spent fuel pool and stored. Due to the unavailability of spent fuel reprocessing capability and the premium placed on space within a facility's spent fuel pool, many LWR's are processing and shipping hardware for disposal shortly after removal from the core. Fuel pool storage space is also the impetus behind fuel rod consolidation projects. Irradiated hardware produced from these projects may add significantly to the quantity of irradiated hardware that must be shipped off site for commercial low-level waste disposal or transferred to the Department of Energy (DOE).

Prior to the shipment of radioactive waste to a disposal facility, federal laws mandate that the radionuclide content of the waste be determined. Low-level radioactive wastes are characterized and classified based on radionuclide limits listed in Section 61.55 Tables of 10 CFR Part 61 (Part 61). The nuclear industry currently uses two different methods to characterize the radionuclide content and distribution in irradiated hardware: (1) the direct sample method, and (2) the activation analysis method. Classification for commercial disposal follows hardware characterization.

Experience indicates that most irradiated hardware is classified as Class C waste. In some cases, the hardware exceeds Class C concentration limits. Hardware classification is driven by the concentration of the relatively short-lived activation product $^{63}$Ni relative to the limit in Table 2 of Section 61.55; and by the combined concentrations of the long-lived activation products $^{59}$Ni, $^{94}$Nb, $^{99}$Tc and $^{14}$C relative to their limits in Table 1 of Section 61.55. These radionuclides are all produced by neutron activation of the base metal constituents which comprise the irradiated hardware. Generally, the concentrations of these radionuclides at the time of shipment depend upon the radionuclide precursor weight fractions in components, the irradiation time and neutron flux, and the elapsed time since components have been removed from neutron exposure.

This report identifies and describes current characterization methods and their implementation. Published data used to characterize hardware radionuclide content and classify hardware for disposal as low-level radioactive waste are also presented. Both characterization methods involve uncertainties. These uncertainties are discussed and, to the extent practicable, bounded in terms of their effect on hardware classification. Suggestions for improving the characterization methods are also presented.

## 2.0 BACKGROUND

Irradiated hardware consists of non-fuel bearing metallic components which are activated by the neutron flux within the reactor core. Requirements for characterizing and classifying these wastes are contained in the NRC regulation 10CFR Part 61 which became effective on December 27, 1982. This regulation requires that all low-level waste be classified according to the concentrations of key radionuclides into one of three classes designated A, B, and C. If the waste material does not exceed Class C limits, it can be disposed at a commercial shallow land burial facility. Where the concentrations of key radionuclides exceed Class C limits, the material can be either disposed at commercial shallow land burial facilities with special permission on a case basis, or, under the provisions of the Low-level Radioactive Waste Policy Amendments Act of 1985, be transferred to DOE for ultimate disposal as greater than Class C waste.

This section describes the various forms of irradiated hardware currently generated by nuclear power plants, their composition, and the radionuclides of concern relative to hardware classification in accordance with Part 61 regulations.

### 2.1 Description of Hardware Components

There are two categories of irradiated hardware: (1) metallic auxiliary core components removed during routine reactor refueling and maintenance, and (2) fuel assembly hardware generated when spent fuel assemblies are disassembled for fuel rod consolidation. For the purposes of this discussion these categories will be referred to as non-fuel assembly components and fuel disassembly components. By far the principal source of radioactivity in both categories of hardware is the radionuclides activated by neutron bombardment of the stable metal constituents of the hardware. All components are primarily constructed of either stainless steel, inconel, Zircaloy, or some combination of these materials. Some non-fuel assembly components also contain neutron absorbing materials such as borosilicate glass, boron carbide or silver-indium-cadmium alloy, or source materials such as antimony-beryllium or californium. In pressurized water reactors (PWR's), these non-fuel assembly components typically consist of reactor control cluster assemblies, burnable poison rod assemblies, source assemblies, thimble plug assemblies, and guide tubes. In boiling water reactors (BWR's), these components typically consist of control rod blades, instrument strings, fuel flow channels, and poison curtains. Both types of reactors also occasionally generate assorted miscellaneous irradiated components when structural components within the reactor vessel are replaced. The fuel disassembly hardware is that portion of a fuel assembly remaining after the fuel pins have been removed during rod consolidation operations. Typically, this hardware includes end fittings, spacer grids, water rods (BWR) or guide tubes (PWR), and assorted springs and fasteners.

### 2.1.1 PWR Non-Fuel Assembly Components

Table 2.1 summarizes the physical characteristics of PWR components. All volumes shown on this Table represent the displaced, as opposed to envelope, volumes of the components. These components are described below.

#### 2.1.1.1 Reactor Control Cluster Assemblies

These components, referred to as RCCA's, are comprised of a Type 304 stainless steel upper section having fairly heavy inconel springs, and a lower section composed of Ag-In-Cd pellets clad in Type 304 stainless steel tubes. During reactor operations, RCCA's are fully withdrawn from the core, and see relatively low neutron fluxes. They are removed from the core and replaced when damaged or when the neutron absorption capability of the Ag-In-Cd is depleted.

#### 2.1.1.2 PWR Thimble Plug Assemblies

These components are comprised of Type 304 stainless steel upper sections having fairly heavy inconel springs and a lower section of short, solid, Type 304 stainless steel rods or plugs. The lower sections of these components are in the active core region during reactor operation and hence are exposed to a significant neutron flux. These components are typically only replaced when damaged and thus have relatively long in-core times.

#### 2.1.1.3 Burnable Poison Rod Assemblies

These components, referred to as BPRA's, are comprised of an upper and a lower section. The upper section is identical to a thimble plug assembly upper section. The lower section is composed of relatively long sections of Type 304 stainless steel annular tubes containing a neutron absorbent material and short, solid, stainless steel rods or plugs. The absorbent material can be boron carbide, cadmium or borosilicate glass. During reactor operations, BPRA's are partially inserted in the core, and see moderate neutron fluxes. They are removed from the core and replaced when damaged or when the neutron absorption capability of the absorber material is depleted.

## Table 2.1 Summary of Typical Hardware Characteristics

| | Overall Length (cm) | Configuration | Areal Cross Section (cmxcm) | Weight (kgs) | Approx. Volume (cc) | Materials of Construction |
|---|---|---|---|---|---|---|
| **PWR Components** | | | | | | |
| Thimble Plug Assemblies | 30 | Square array of plugs | 15x15 | 5.5 | 570 | 304 SS/Inconel |
| Burnable Poison Assemblies | 400 | Square array of plugs and rods | ---- | 13.6 to 20.4 | 3700 to 5100 | 304 SS/Inconel/ Borosilicate |
| Control Rod Assembiles | 400 | Square array of plugs and rods | ---- | 74.8 | 8500 | 304 SS/Inconel/ Ag/In/Cd |
| Source Assemblies | 400 | Square array of plugs and rods | ---- | 13.6 to 16 | 4300 | 304 SS/Inconel/ Borosilicate and Source Material |
| Thimble Plug Guide Tubes | variable | Tubing | 1 cm diam. | N/A | N/A | 304 SS |
| **BWR Components** | | | | | | |
| Control Rod Blades | 440 | Cruciform | 25x25 | 130 | 18000 (.018m$^3$) | 304 SS and Boron Carbide |
| Poison Curtains | 400 | Flat plate | 21.0x.2 | 16.4 | 1700 | Borated SS |
| Flow Channels | 420 | Rectangular box | 14x14 | 30 | 4800 | Zircaloy/304 SS |
| Instrument Strings | variable | Tubing | 1.9 cm diam. | N/A | N/A | 304 SS |

References: 2, 30, 31, 32

#### 2.1.1.4 Primary and Secondary Source Assemblies

The configuration of these assemblies is identical to BPRA's except that in one or two of the long Type 304 stainless steel tubes the neutron absorber material is replaced by a primary or secondary neutron source material. The primary source assemblies contain californium or some other neutron source material, while the secondary source assemblies typically contain antimony-beryllium. The secondary sources can be shipped with the components for disposal.

#### 2.1.1.5 In-Core Instrument Guide Tubes

These components are long sections of Type 304 stainless steel tubing through which instruments are inserted in the core. Since a portion of the component resides within the core region, the component contains both a "hot" and a "cold" end. These instrument tubes sometimes include fission chambers which contain transuranic material. Fission chambers are constructed differently than other irradiated hardware components and the methods described herein are not used to characterize their transuranic content. They can, however, be disposed at commercial sites provided that Class C limits for transuranics are met.

### 2.1.2 BWR Non-Fuel Assembly Components

Table 2.1 also summarizes the physical characteristics of BWR components. The volumes shown on this Table are the displaced volumes of the components. These components are described below.

#### 2.1.2.1 Control Rod Blades

These components, referred to as CRB's, are constructed primarily of Type 304 stainless steel. They consist of a stainless steel upper handle followed by a long cruciform section composed of stainless steel tubes containing boron carbide pellets and clad in a stainless steel sheath, followed by a solid stainless steel velocity limiter section. In some of the older CRB designs, the boron-containing tubes were made of inconel-600. The upper section of the cruciform also contains stellite rollers which are usually removed prior to disposal. More recent CRB designs incorporate pins and rollers made of low cobalt content materials. These components are inserted from the bottom of the core and while most of the CRB is not within the core region, the upper sections are exposed to relatively high neutron fluxes.

#### 2.1.2.2 SRM, IRM and LPRM Instrument Strings

These components are long sections of Type 304 stainless steel tubing through which instruments are inserted in the core. Since a portion of the component resides within the core region, the component contains both a "hot" and "cold" end. These instrument tubes sometimes include fission chambers which contain transuranic material, and the methods described herein are not used to characterize their transuranic content. They can, however, be disposed at commercial sites provided that Class C limits for transuranics are met.

#### 2.1.2.3 Poison Curtains

These components consist of long, thin, flat plates of borated Type 304 stainless steel having a stainless steel handle. They are used to flatten the flux during initial startup. Thus, they are within the core region for the first cycle and are exposed to a relatively high neutron flux.

#### 2.1.2.4 Fuel Flow Channels (Fuel Shroud)

These components are long rectangular boxes which surround the fuel assemblies. They are generally constructed of either Zircaloy-2 or Zircaloy-4 although some older BWR's have shrouds of Type 304 stainless steel.

### 2.1.3 Fuel Disassembly Hardware

Characteristics of disassembly hardware are summarized in Table 2.2. These components generally consist of Type 304 stainless steel end fittings; Zircaloy-4 and inconel-718 spacer grids; and assorted 316L stainless steel, 304L stainless steel, and inconel-718 springs and fasteners. Since all disassembly components are exposed to a significant neutron flux, they are highly activated.

## 2.2 Irradiated Hardware Classification

As low-level radioactive waste, irradiated hardware must be classified according to the requirements of Part 61 prior to disposal. These regulations provide concentration limits for certain isotopes in radioactive wastes, and based on these limits divide the waste into three classes designated A, B and C. The radionuclides identified in Part 61 which drive the classification of irradiated hardware are presented in Table 2.3, a partial reproduction of Tables 1 and 2 of Section 61.55.

These radionuclides are all activation products having relatively long half-lives. Some of the radionuclides in this table do not emit measurable gamma radiation in their decay schemes and are measured only through complicated and time consuming radiochemical and nuclear counting processes. The

Table 2.2 Summary of Typical Spent Fuel Disassembly Hardware Characteristics

| | Overall Length (cm) | Configuration | Areal Cross Section (cmxcm) | Weight (kgs) | Approx. Volume (cc) | Materials of Construction |
|---|---|---|---|---|---|---|
| **PWR Components** | | | | | | |
| End Fittings | 6 to 30 | Square | 21x21 | 11.3 to 18.1 | 1400 to 2250 | 304 SS |
| Guide Tubes | variable | Tubing | 1 cm diam. | 7.5 to 21.2 | NA | Zircaloy |
| Spacer Grids | 7 | Square | 21.21 | 4.0 to 12.6 | 525 to 1660 | Inconel/Zircaloy |
| Miscellaneous (Springs, etc.) | ----- | ----- | ----- | 0.7 to 11.4 | NA | 304 SS/Inconel |
| **BWR Components** | | | | | | |
| End Fittings | 17 | Square | 15x15 | 5.4 to 15.2 | 680 to 1900 | 304 SS |
| Guide Tubes | variable | Tubing | 1 cm diam. | 1.7 to 13.2 | NA | Zircaloy |
| Spacer Grids | 4 to 6 | Square | 15x15 | 1.1 to 1.6 | 145 to 210 | Inconel/Zircaloy |
| Miscellaneous (Springs, etc.) | ----- | ----- | ----- | 0.2 to 1.1 | NA | Inconel |

Reference: 11

Table 2.3 10 CFR Part 61 Limits For Class C Low-level Radioactive Waste[a]

Table 1 Long-Lived Radionuclides

| Nuclide (half-life) | Concentration (curies/m^3) |
|---|---|
| Carbon-14 in activated metal(5,730 yrs) | 80 |
| Nickel-59 in activated metal(75,000 yrs) | 220 |
| Niobium-94 in activated metal(20,000 yrs) | 0.2 |
| Technetium-99 (214,000 yrs) | 3 |

Table 2 Short-Lived Radionuclides

| Nuclide (half-life) | Concentration (curies/m^3) |
|---|---|
| Nickel-63 in activated metal (100 yrs) | 7,000 |

---

Note

a. Limits are for single radionuclides; limits for mixtures of radionuclides are determined by a sum-of-fractions rule separately for long-lived and for short-lived radionuclides. The sum of fractions for either short or long-lived radionuclides is determined by dividing each nuclide's concentration by its Class C limit and adding the resulting values. If the sum exceeds 1 for either short or long-lived radionuclides, the waste is greater-than-Class-C.

NRC regulations recognize that radionuclide detection may be difficult, and allow licensees to determine the concentration of hard-to-detect radionuclides by "scaling" from measurable radionuclides. For irradiated hardware, the radionuclides shown on Table 2.3, as activation products, are scaled from $^{60}$Co.

As described above, all hardware components are composed of stainless steel, inconel, Zircaloy, or some combination of these materials. The radionuclides of concern for classification are $^{14}$C, $^{59}$Ni, $^{94}$Nb, and $^{99}$Tc from Table 1, and $^{63}$Ni from Table 2. Other radionuclides of interest, not specifically identified in the Part 61 Tables, except as significant radionuclides with half lives less than 5 years, are $^{54}$Mn, a gamma emitter,and $^{55}$Fe, a beta emitter. For some components other significant gamma emitters may also be present. These include $^{125}$Sb in BWR flow channels and $^{110m}$Ag in PWR reactor control cluster assemblies.

These radionuclides, plus $^{60}$Co, the dominant gamma emitting radionuclide in hardware, are the principal radionuclides for classification under Part 61. This does not mean that other radionuclides listed in Part 61 are not present. The exposure of components to reactor coolant and storage in the spent fuel pool result in surficial coatings of transuranics and fission product radionuclides such as $^{90}$Sr and $^{137}$Cs. The presence of these radionuclides should have little effect on the classification of irradiated hardware.

The precursors for the above activation products are found in the base metals used for the components. Table 2.4 shows ranges of precursor compositions in stainless steel, Zircaloy and inconel. As shown, nickel dominates in inconel, is a major constituent in stainless steel, and is present in trace quantities in Zircaloy. Iron dominates in stainless steels, and is present in the inconels. Conversely, cobalt, niobium, molybdenum, and nitrogen are trace constituents in all component base metals. The mechanisms which lead to production of the radionuclides of concern are discussed below.

### 2.2.1 Cobalt-60

Cobalt-60 is the most important radionuclide in the classification and characterization of irradiated hardware. All of the characterization methodologies analyzed by this study first estimate the concentration of $^{60}$Co in the irradiated hardware, and subsequently scale the concentrations of the other activation products to it.

Cobalt-60 is produced principally by (n,gamma) activation of trace concentrations of natural cobalt found in stainless steel, inconel, and Zircaloy. Cobalt-60 contributes significantly to the total curie content and dose rates for

## 2.4 Summary of Typical Composition Ranges For Metals Used in Hardware[1]

| Element | Stainless Steels | | Inconels | | Zircaloy |
|---|---|---|---|---|---|
| | 304 | 316 | 600 | 700 | |
| Fe (%) | 68-76 | 64-65 | 6-10 | 7-18 | 1.5-2.2 |
| Ni (%) | 8.8-11 | 12.5-13.2 | 58-72 | 52-72 | .02-0.5 |
| Co (ppm) | 750-2600 | 1300-1600 | 400-700 | 400-700 | 10 |
| N (ppm) | 400-520 | 300-400 | 1000-2000 | 1000-2000 | 60-100 |
| Nb (ppm) | 10-200 | 10-65 | 10-400 | 10-400 | 100-150 |
| Mo (ppm) | 1900-5500 | 2000-2200 | 2000-3000 | 2000-3000 | 0 |

Notes

1. Ranges shown were obtained from a variety of sources.

irradiated hardware. For most irradiated components, gamma radiation from the decay of $^{60}$Co represents greater than 95% of the total measured dose rate. Exceptions are BWR flow channels and PWR reactor cluster control assemblies.

### 2.2.2 Carbon-14

Carbon-14 is produced in irradiated hardware primarily by the (n,p) reaction on trace concentrations of nitrogen found in stainless steel and inconel. Carbon-14 usually does not contribute significantly to the total curie content, but it contributes to the Table 1 sum of fractions.

### 2.2.3 Nickel-59

Nickel-59 is one of the most important activation products, with an extremely long half-life. Nickel-59 is produced primarily by (n,gamma) activation of naturally occurring $^{58}$Ni. Natural nickel represents about 8 to 11 w/o in Type 304 stainless steels, and 52 to 72 w/o in inconels. This particular radionuclide does not contribute significantly to the total curie content but is a major contributor to the Table 1 sum of fractions.

### 2.2.4 Niobium-94

Niobium is found as a trace contaminant in stainless steels, inconels, and Zircaloy. The radionuclide $^{94}$Nb is produced by (n,gamma) reactions on natural niobium. It does not contribute significantly to the total curie content, but affects classification as a major contributor to the Table 1 sum of fractions.

### 2.2.5 Technetium-99

Technetium-99 is produced as an activation product from the (n,gamma) reaction and subsequent beta decay of natural molybdenum which is a trace contaminant. It does not contribute significantly to the total curie content, but its presence affects the Table 1 sum of fractions.

### 2.2.6 Nickel-63

Nickel-63 is one of the most abundant activation products expected from hardware irradiation. It is created primarily from the (n,gamma) reaction on naturally occurring nickel, which is present in inconel in the 52 to 72 w/o range, and in Type 304 stainless steel in the 8 to 11 w/o range. Nickel 63 is not a major contributor to the total curie content of a component.

### 2.2.7 Manganese-54

Manganese-54 is produced from (n,p) reactions on the stable iron isotope $^{54}$Fe. Manganese-54 can be a contributor to the total curie content but does not affect component classification. However, $^{54}$Mn also has a relatively high energy gamma decay, and can contribute 5% or more to the total measured dose rate for components just removed from the core.

### 2.2.8 Iron-55

Iron-55 is produced from the (n,gamma) reaction on natural iron which is present in stainless steel in the 64.0 to 76.0 w/o range, and in inconel in the 6.0 to 18.0 w/o range. Iron-55 does not impact the classification of irradiated hardware. However, $^{55}$Fe represents a large fraction of the total curie content for irradiated stainless steels. Even with its 2.6 year half life, significant quantities of $^{55}$Fe will be present in irradiated hardware several years after removal from the core. The presence of $^{55}$Fe will thus impact the disposal costs of irradiated hardware.

### 2.2.9 Antimony-125

Antimony-125 is produced from the (n,gamma) reaction and subsequent beta decay of natural tin. Antimony-125 is found in irradiated Zircaloy components, which contain between 1 and 2 w/o tin. With a half-life of 2.7 years, $^{125}$Sb may represent a significant fraction of the total curie content, and measured gamma dose rate, of irradiated Zircaloy shortly after removal from the neutron flux. However, it will generally not affect component classification.

### 2.2.10 Silver-110m

Silver-110m is produced as an activation product from the (n,gamma) reaction of natural silver found in those components containing Ag-In-Cd alloy neutron absorbing material. Similar to $^{125}$Sb, the $^{110m}$Ag produced will not affect classification, but may represent a significant fraction of the total curie content and gamma dose rate, measured shortly after removal from the neutron flux, for components containing Ag-In-Cd. Silver-110m has a half-life less than one year.

## 3.0 REVIEW OF CURRENT CHARACTERIZATION AND CLASSIFICATION METHODS

Two methods are used to characterize and classify irradiated hardware: the direct sample method and the activation analysis method. The direct sample method is used by Science Applications International Corporation (SAIC). Variations of the activation analysis method are used by Waste Management Group, Inc. (WMG), Torrey Pines Technology (TPT) and Battelle Pacific Northwest Laboratories (PNL), as well as several utilities.

This section first discusses the sources of hardware radioactivity, then provides a brief description of each of the characterization techniques. Detailed descriptions of the techniques are presented in Appendix A.

### 3.1 Activation of Hardware

The hardware characterization methods used by vendors and utilities initially determine the $^{60}$Co content at the time of measurement. Then, using scaling factors for Part 61 radionuclides, the methods determine the Part 61 radionuclide content and classify the component.

The metal constituents of primary concern in hardware are nickel, cobalt, nitrogen, molybdenum, niobium, and iron. These precursor elements are activated due to bombardment by neutrons in the reactor core. The concentration of the Part 61 radionuclides at the time of characterization depends primarily on the initial precursor composition in the base metal, the neutron flux, the time of neutron exposure, and the decay time after removal from the core.

The interaction of these effects on Part 61 radionuclide concentrations can be illustrated using three key radionuclides. These radionuclides are: $^{60}$Co because it is the base radionuclide used for scaling Part 61 radionuclides, $^{55}$Fe because it is a major constituent and has a short half life, and $^{63}$Ni because it is a major Part 61 radionuclide affecting classification. A Type 304 stainless steel component having an initial composition of 0.1 w/o cobalt, 10 w/o nickel, and 68 w/o iron was considered typical.

Figure 3.1 shows the concentration of $^{55}$Fe relative to $^{60}$Co as a function of irradiation time. Figure 3.2 shows similar data for $^{63}$Ni. As shown the relationships between the concentrations of these radionuclides and $^{60}$Co, the basis for determining scaling factors, vary considerably. For $^{55}$Fe, the difference between a 3 and 5 year irradiation time represents about an 11 percent decrease in the scaling factor. For $^{63}$Ni, the same time frame represents an increase of

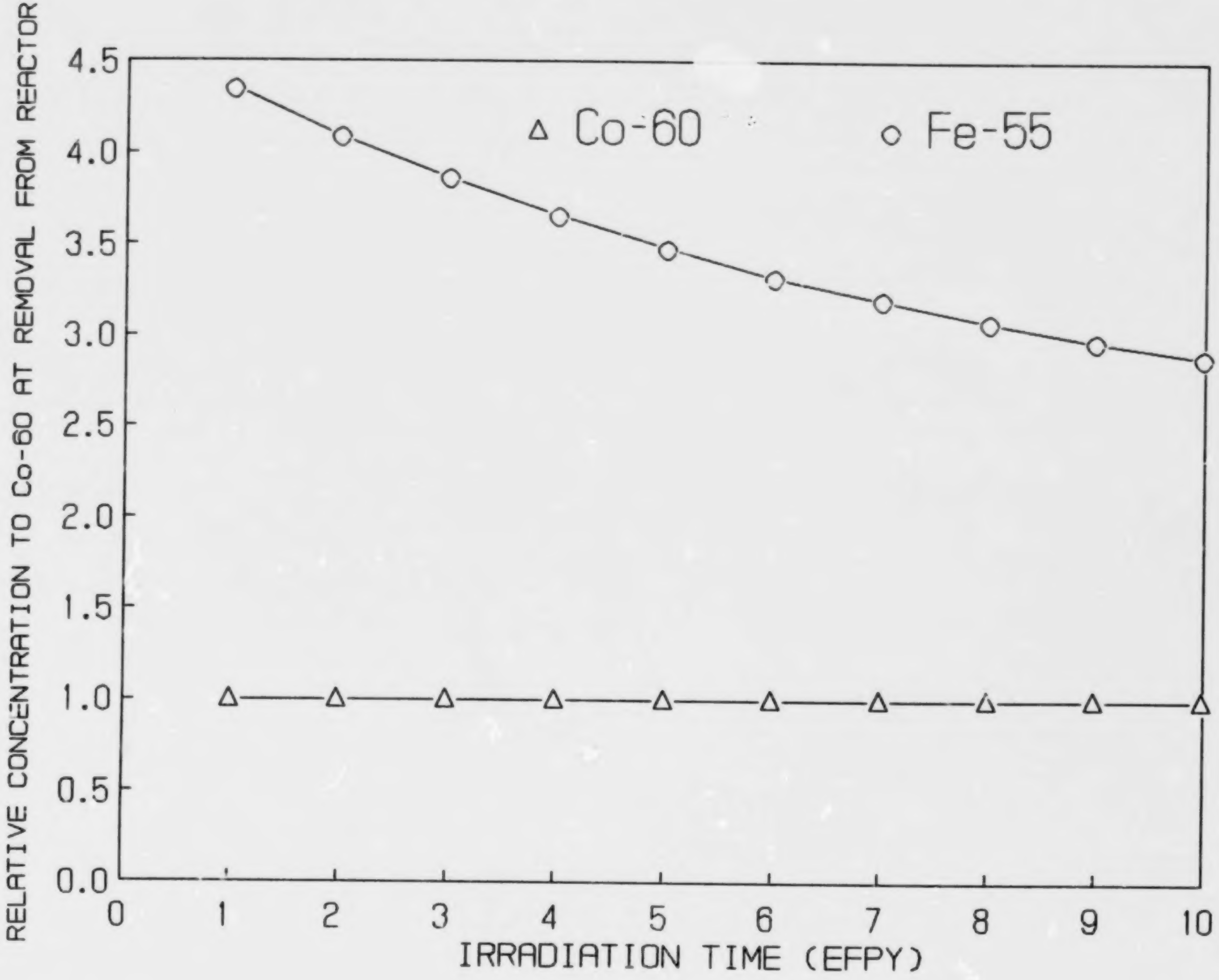

**Figure 3.1** $^{55}$Fe Concentration Relative to $^{60}$Co

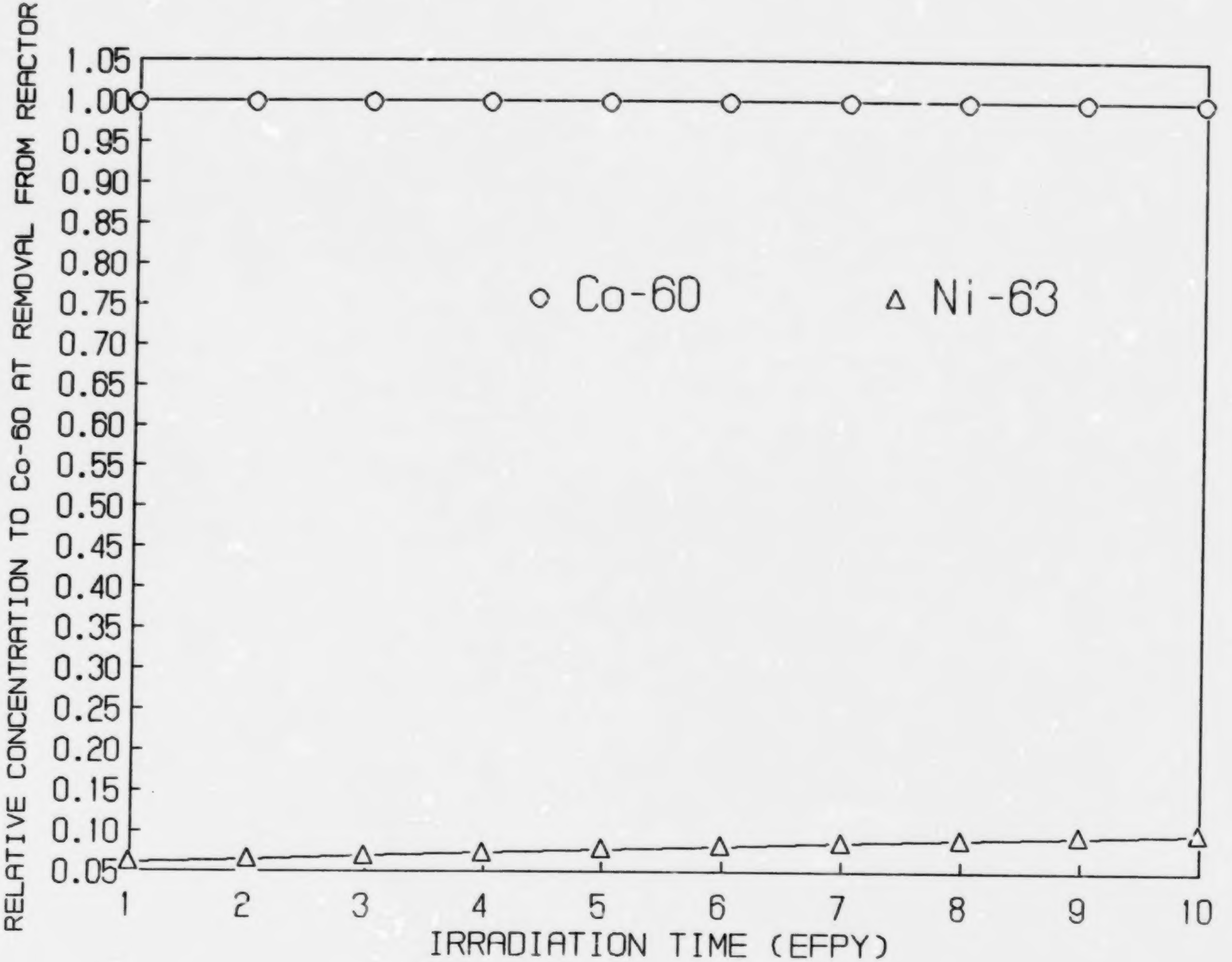

Figure 3.2 $^{63}$Ni Concentration Relative to $^{60}$Co

about 10 percent in the scaling factor. When the component is removed from the core, the base metals are no longer activated and the Part 61 radionuclides present at discharge continue to decay.

The effect of decay time on scaling factors is illustrated by Figures 3.3 and 3.4. These figures show the concentrations of $^{55}$Fe and $^{63}$Ni relative to $^{60}$Co as a function of time after core removal using the 5-year irradiation time concentrations from Figures 3.1 and 3.2. As shown on Figure 3.3, the difference between a 3 and 5 year cooling time represents about a 33 percent decrease in the $^{55}$Fe scaling factor. For the relatively long lived $^{63}$Ni, the same two year difference in decay time results in about a 27 percent increase in the scaling factor. Since the only radionuclide concentration determined at the time of characterization is $^{60}$Co, both irradiation time and cooling time are important factors in hardware characterization.

## 3.2 Surface Contamination of Hardware

In addition to activation product radionuclides, hardware components are surficially contaminated by both fission product and transuranic radionuclides. Principal fission product radionuclides include $^{137}$Cs and $^{90}$Sr. Principal transuranic radionuclides include the beta emitters ($^{241}$Pu and $^{242}$Cm) and the alpha emitters. These radionuclides are present in reactor core coolant and in the spent fuel pool, and to some degree contaminate the hardware. The extent to which these surface contaminants are deposited and, in turn, affect waste classification depends on the following factors:

1. To what extent are these contaminants present in the reactor coolant?
2. What mechanisms lead to their deposition on hardware while the hardware is exposed to reactor coolant?
3. To what extent are these contaminants present in the spent fuel pool?
4. What mechanisms lead to their deposition on hardware while the hardware is in the fuel pool?
5. To what extent can these radionuclides be deposited on hardware in both the reactor vessel and the spent fuel pool?
6. What is the volume/weight of deposition on an absolute basis and relative to the displaced volume/weight of the component being classified?

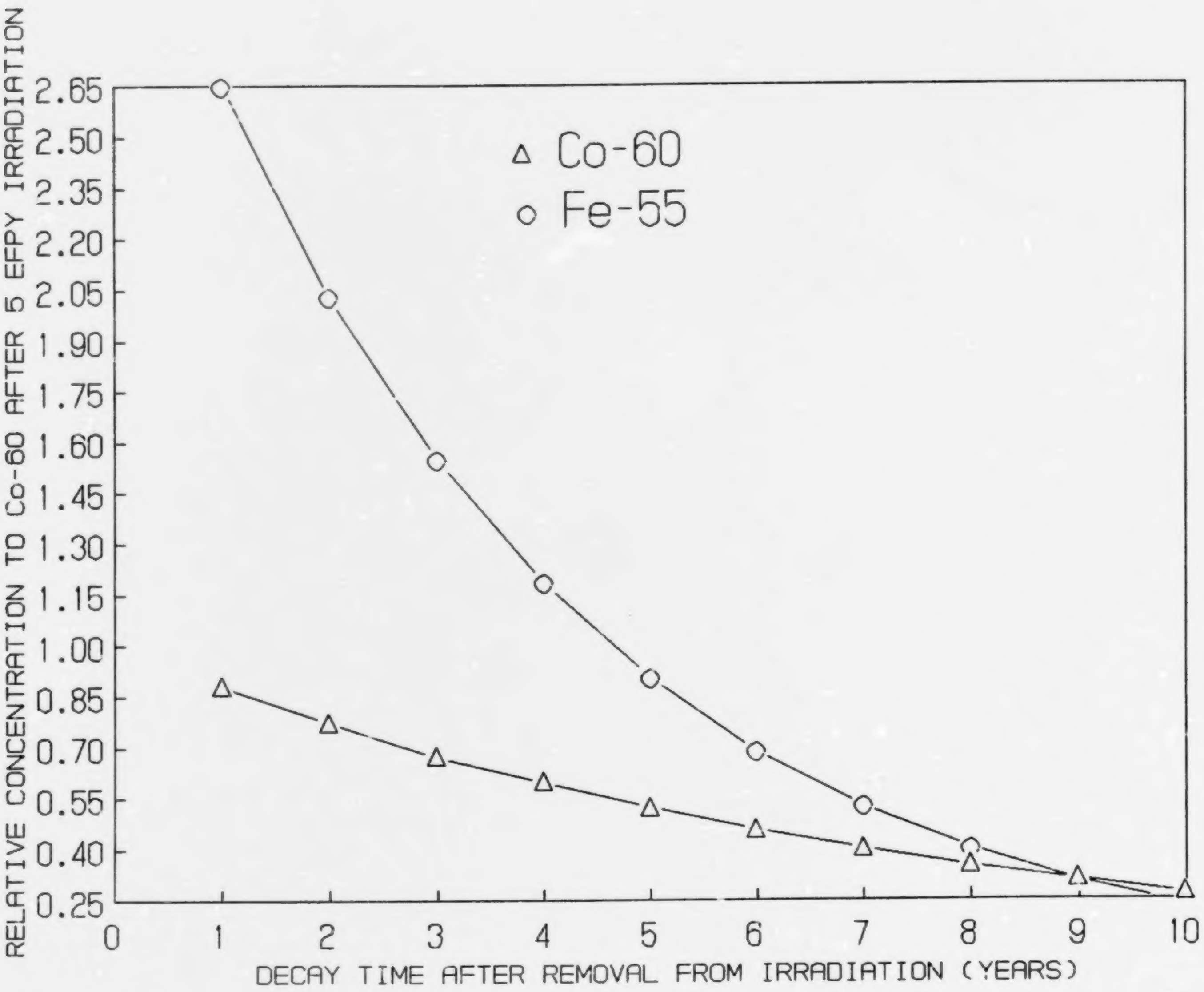

**Figure 3.3** Relative Decay of $^{60}$Co And $^{55}$Fe

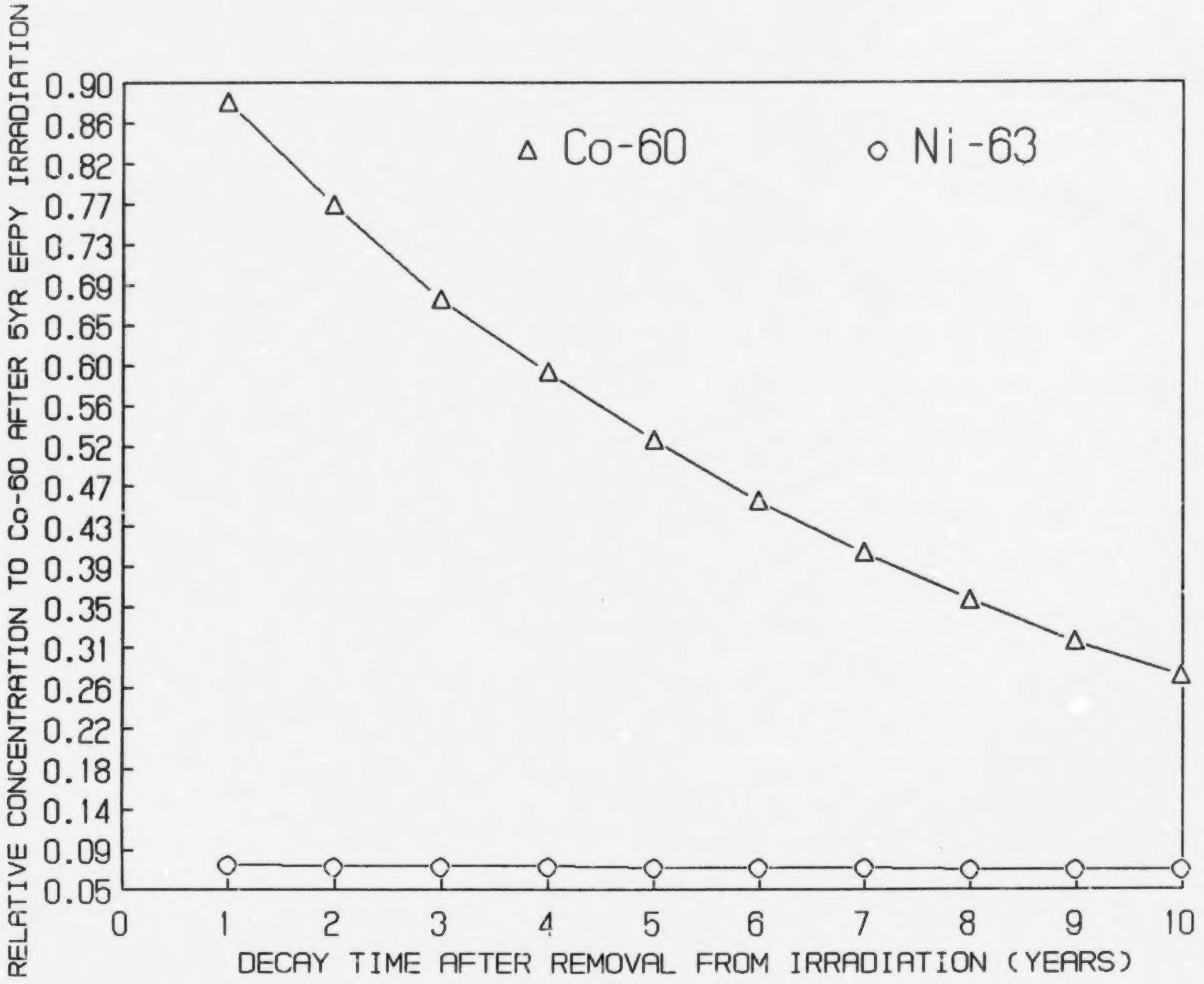

**Figure 3.4** Relative Decay of $^{60}$Co and $^{63}$Ni

7. To what degree does this surficial deposition of Table 1 radionuclides (transuranics) and Table 2 radionuclides ($^{137}$Cs and $^{90}$Sr) impact application of the sum of the fractions rule for waste classification?

Since both sources of contamination arise from fuel leaks or failures, the presence of these contaminants in the reactor core should be initially examined. In this environment, the hardware is in a high flow regime, the quality of the water is carefully maintained to minimize contaminants, and conditions such as high temperatures are present which could promote deposition on hardware. Leaking fuel assemblies are typically removed as soon as practicable and placed in the spent fuel pool.

The environment in the spent fuel pool is somewhat different. The hardware is stored with other components, including fuel assemblies, under low flow and relatively low temperature conditions. In addition, fuel pool water is not maintained to the same quality as reactor coolant. Experience with this environment shows that crud builds up on components with time, and that movement of components in most pools results in re-distribution and deposition of this crud.

Considering the conditions at most fuel pools and experience with hardware removed from the pools, particularly fuel racks, crud tends to be deposited in crevices and on horizontal flat surfaces. Radiation surveys of racks and swipe samples confirm this distribution for gamma emitters and there is no reason to expect any differences for other fission products and transuranics. The exact extent of deposition on a given component can only be determined by swiping all surfaces of the component and statistically analyzing the swipe sample results - an impractical solution. The point, however, is that deposition on components is not uniform.

The relative deposition volume is thus a variable that can be either estimated or measured directly for selected components. Components where surface deposition could have the most impact include thin-walled tubing used for guide tubes and instrument strings. A component of this type will typically have a surface-to-volume ratio of 10-to-1 or more.

To illustrate, consider a stainless steel instrument tube (LPRM) 144" long, having an outside diameter of 0.75" and a wall thickness of 35 mils, a surface area of 650 in$^2$, a displacement volume of 11 in$^3$, and a weight of approximately 1500 grams. If fuel pool silt containing 10 nCi/gm TRU covers the surface of the tube to a thickness of 2 mils, the tube will have a total of approximately 20 cm$^3$ of TRU contaminated material deposited on the surface. Based on a conservative silt density of 2 $gm/cm^3$, the tube surface will total 400 nCi of TRU isotopes, translating to an overall concentration of 0.3 nCi/gm TRU for the component. Surface samples obtained from LPRM components have been radiochemically analyzed by SAIC. These sample results show TRU concentrations ranging between 1.4 nCi/gm and 9.0 nCi/gm(1).

These results would indicate that component surface TRU contamination should generally have no effect on Part 61 classification. However, since disposal sites require reporting of transuranic content, hardware paperwork should include transuranic information.

### 3.3 SAIC Quantiscan(tm) Method

The SAIC direct sample method involves the application of gamma scanning, direct bulk and surface sampling of selected components, and radiochemical analysis of samples. It is based on the following assumptions:

1. The elemental constituents of metal components from the same batch (i.e. constructed at the same time) are constant throughout the batch.
2. The activity seen in the scanner collimator represents the average concentration contributing to a contact dose rate measured at the same location.
3. The material in the direct sample is either well mixed, or sees the same neutron flux as the rest of the component.
4. Fission product and TRU activities exist only in surface deposits (except for fission chambers in monitors).
5. Surface deposits were deposited while the component was in the reactor vessel, and the relative radionuclide makeup of fission product and TRU activity is constant along the surface.
6. Crud radionuclide concentrations are proportional to the bulk concentrations in the component (i.e., crud deposition is a function of neutron flux and temperature).

Hardware is characterized in two distinct phases. The first phase is performed by SAIC and involves characterization of representative components using a combination of gamma spectroscopy, direct sampling and contact radiation profiling. The second phase is performed by a separate disposal service contractor. During phase two, $^{60}$Co dose-to-curie conversion factors and scaling factors determined during phase one are used in conjunction with contact radiation profiles to characterize the bulk of the hardware to be shipped. The elapsed time between these two phases can typically vary from about 6 weeks to about 6 months.

The first phase of the Quantiscan (tm) characterization process involves the following steps:

1. Using the Quantiscan (tm) device, determine the $^{60}$Co content of the component at a point location.

2. Using an RO-7 ionization chamber, obtain a contact radiation reading at the same location.

3. Calculate a $^{60}$Co dose-to-curie conversion factor from the gamma scan input and the contact radiation level reading at the scan location.

4. Obtain a bulk metal sample at the scan location.

5. Obtain a surface contamination sample for the component at a location near that for the bulk metal sample.

6. Radiochemically analyze the bulk metal sample to determine the relative content of Part 61 radionuclides and $^{60}$Co and thus determine scaling factors.

7. Radiochemically analyze the surface sample to determine scaling factors used to estimate fission product and transuranic surface contamination.

8. Obtain contact radiation profiles over the axial length of the sampled component using an RO-7 ionization chamber.

9. Apply the $^{60}$Co dose-to-curie conversion factor to the axial radiation level readings to obtain the $^{60}$Co content of the component.

10. Use the direct sample scaling factors and the profile based $^{60}$Co activities to determine the activities of the Part 61 radionuclides.

11. Use the above profiles and the results of the surface contamination sample to estimate the extent of component surface contamination.

12. Compare the Part 61 radionuclide concentrations for the sampled component to the Part 61 classification limits, and classify the component.

The results of this phase are presented in a report prepared by SAIC. The steps involved in the second phase of a typical project include the following:

1. Using an RO-7 ionization chamber, perform contact radiation profile measurements for components to be shipped.

2. Characterize components using the component specific $^{60}$Co dose-to-curie conversion factor and scaling factor sets obtained during phase one.

3. Process components for volume reduction and shipment.

The vendor performing phase two usually processes the hardware and packages it for disposal. The information needed to classify the hardware and prepare shipping papers is also produced by the phase two vendor.

## 3.4 Activation Analysis Methods

Activation analysis methods are used by several vendors and utilities. The specific techniques used to apply activation analysis differ among organizations. These techniques are summarized below.

### 3.4.1 WMG RADCOR (tm) Method

RADCOR (tm)(2) is a computer code which characterizes irradiated hardware using a form of activation analysis. It is based upon the following assumptions:

1. The parameters which define the physical and chemical characteristics of the hardware can be defined in a component specific data base.

2. The irradiation history and cooling time for hardware components are available.

3. Radiation level measurements are available.

4. There is a direct correlation between $^{60}$Co content and radiation measurements.

5. The distribution of Part 61 radionuclides is a function of the chemical constituents in the hardware before irradiation, the irradiation history, and the elapsed time since removal from the core.

A typical project involves two phases. The first phase develops a component specific data base which defines the physical characteristics, the irradiation history, and the cooling times for each batch of components to be characterized. The second phase is performed by WMG in conjunction with a disposal service field contractor with WMG preparing shipment paperwork and other documentation. It involves use of phase one component data and non-contact radiation profiles to characterize each component. The elapsed time between these phases is typically several weeks.

The RADCOR (tm) characterization and classification process involves the following steps:

1. Define a project component specific data base.
2. Determine component specific $^{60}$Co dose rate-to-curie conversion factors using the MICROSHIELD (tm)(3) computer code.
3. Obtain a sample of the fuel pool silt to estimate the extent of transuranic contamination on component surfaces.
4. Define component specific radiation level breakpoints to be applied during volume reduction processing.
5. Obtain radiation profiles for all components over their axial length, at 6 inches, using an RO-7 ionization chamber.
6. Characterize each component:
   - Determine the $^{60}$Co content of the component at the time of radiation measurement,
   - Decay correct the $^{60}$Co content back to the time of component removal from the reactor core.
   - Calculate scaling factors based upon the activation of the Part 61 radionuclides relative to the activation of $^{60}$Co.
   - Calculate the Part 61 radionuclide activities at the time of removal.
   - Decay all radionuclide activities to the time of characterization and classification.
7. Classify each component using the methodologies in the NRC approved RADMAN(4) computer code.
8. Transfer each component record to a historic archive file.

9. During packaging, determine DOT and NRC classification over all components placed within the disposable container.

10. Prepare shipping papers, Class C waste documentation, and cask decay heat report.

11. Archive disposable container records including individual component content.

Upon completion, each project is documented with traceability on all inputs and components.

### 3.4.2 Torrey Pines Technology Method

The Torrey Pines Technology (TPT) technique is similar to RADCOR(tm). The TPT technique uses a form of activation analysis coupled with radiation measurements to characterize hardware. Differences between the TPT and RADCOR(tm) techniques include:

1. Component-specific $^{60}$Co dose rate-to-curie conversion factors are calculated using the PATH(5) computer code. RADCOR(tm) uses the MICROSHIELD (tm) computer code.

2. A two-region neutron flux spectrum computer code, MICROX(6), is used to calculate spectrum-dependent cross sections. RADCOR(tm) relies on published cross section data.

3. Component radiation profiling for similar components is performed somewhat differently in terms of the number of axial measurements taken.

TPT uses non-contact component specific radiation profiles to characterize each component. A typical project is documented in two parts: an initial report describing the basis for application of the technique, and a final report with component and disposable container specific radionuclide content.

### 3.4.3 Battelle Pacific Northwest Laboratories Method

The Battelle Pacific Northwest Laboratories (PNL) technique uses activation analysis to calculate long-lived activation products in reactor construction material for the purpose of reactor decommissioning. The PNL methodology has been formally published as NUREG/CR-3474(7).

PNL has performed a characterization and classification study for a BWR facility(8) which employed scaling factors generated from the NUREG/CR-3474 study. In this project, transuranic contents for local power range monitors (LPRM's) were calculated using activation analysis algorithms found in the ORIGEN-2(9) computer code. This analysis used the initial uranium isotopic concentrations as provided by the manufacturers of the LPRM's and flux data as provided by the utility. This study also characterized CRB's.

Other Part 61 radionuclide concentrations were measured in the following manner:

1. Determine the $^{60}$Co concentration from a measurement of the gamma exposure rate at a location on a component.

2. Calculate the $^{60}$Co dose rate-to-curie conversion factor at the above scan location using the SLAB(10) gamma shielding computer code.

3. Obtain contact radiation profiles over the axial length of selected components using thermo-luminescent dosimeters (TLD).

4. Using published correlation factors from NUREG/CR-3474 and other data bases, calculate the concentrations of the balance of the Part 61 radionuclides at all axial locations.

PNL also performed a study to define repository disposal requirements for spent fuel disassembly hardware and other non-fuel hardware components. This study(11) exclusively used the ORIGEN-2 computer code to calculate Part 61 radionuclides for fuel assembly designs and other in-core hardware.

### 3.4.4 Utility Experience

Several utilities, with operating BWR's and PWR's, have performed characterization and classification projects. These projects have generally been performed using the activation analysis algorithms found in ORIGEN-2 or similar codes. This characterization technique does not employ the use of scaling factors to calculate the concentration of Part 61 radionuclides; their concentrations are calculated directly by ORIGEN-2.

The ORIGEN-2 calculations are then benchmarked against measured underwater dose rates for the component of interest, and against a dose rate calculated with a shielding code using the ORIGEN-2 gamma flux output information as input. Adjustments are made as appropriate to bring nuclide concentrations in line with the measured dose rates.

## 4.0 HARDWARE CHARACTERIZATION DATABASE

To document information on hardware characteristics and to compare characterization method results, WMG established a computerized data base using the LOTUS 1-2-3 (tm)(12) software. The reports used to compile the data base are referenced in Section 7.

The frame work established for the data base consists of method and component specific information. The information in the data base framework includes:

1. component type,
2. characterization method,
3. plant type,
4. irradiation history in equivalent full power years,
5. core removal date,
6. characterization date,
7. displaced volume of component,
8. $^{60}$Co dose-to-curie conversion factors,
9. Part 61 radionuclide concentrations,
10. Part 61 radionuclide scaling factors.

The extent to which this information was available varied with method and among the different vendors applying the activation analysis method. The information in the data base represents 12 BWR and 3 PWR characterization projects. It describes 74 distinctly different hardware components from these 15 projects. This information includes characteristics for the following types of components:

BWR Hardware
- 22 control rod blades,
- 22 instrument strings (LPRM, SRM, IRM),
- 6 fuel flow channels,
- 3 poison curtains.

PWR Hardware
- 4 thimble plug assemblies,
- 3 reactor control cluster assemblies,
- 2 instrument strings (Guide Tubes),
- 3 source assemblies,
- 9 burnable poison rod assemblies.

The information for BWR's was obtained from characterization projects using both the direct sample and activation analysis methods. PWR information was only available from projects using the activation analysis method.

Activation analysis input information is more extensive than that currently used for direct sampling. Generally, information missing from direct sample projects included the exposure history in effective full power years (EFPY) and component core removal dates.

The information in the data base was sorted to define the ranges of parameters determined by each method, to compare method results to the extent practicable, and to compare the results for the different activation analysis techniques. The results of these data base reviews and the general conclusions drawn from available component data are discussed below.

## 4.1 General Hardware Characteristics

Table 4.1 is a partial listing of the data base. It shows the descriptive information for the component, including volumes, reported $^{60}$Co dose-to-curie conversion factors, and $^{60}$Co content. Guidelines for review of this table are as follows:

1. Data was included if it represented the same type of component for different plants, or the same types of components with different histories or scaling factors for the same plant,
2. The $^{60}$Co dose-to-curie conversion factors reported by SAIC apply to contact radiation level readings; the conversion factor values for other methods apply to 6-inch radiation level readings,
3. $^{60}$Co dose-to-curie conversion factors reported in units of Ci per $m^3$ per R/Hr were converted to Ci per R/Hr by using the reported component displacement volumes in $m^3$,
4. Reported $^{60}$Co contents for direct sample method results are generally lower than for other methods because the sampled component must have relatively low concentrations to permit radiochemical analysis of the bulk sample.

General observations concerning the information in Table 4.1 are as follows:

1. The amount of available data is sparse relative to the number of components characterized and shipped from the 15 plants included in the data base.

## Table 4.1 Hardware Characteristic Summary

| HARDWARE KIND | HARDWARE VOLUME (M**3) | METHOD | PLANT TYPE | CHARCTERIZATION DATE | DOSE CURIE CO-60 CONV. (CI/R/HR) | DOSE CURIE CO-60 CONV. (CI/M^^3/R/HR) | Co-60 (Ci) | (CI/M^^3) CO-60 |
|---|---|---|---|---|---|---|---|---|
| CRB-12-1# | 1.514E-02 | SAIC | BWR | | 3.028E-02 | 2.000E+00 | NA | NA |
| CRB-4-13035 | 1.514E-02 | SAIC | BWR | 05/06/1986 | 3.179E-02 | 2.100E+00 | 2.120E+02 | 1.400E+04 |
| CRB-1-1 | 1.200E-02 | SAIC | BWR | <04/22/1985 | 4.000E-02 | 3.333E+00 | 6.000E+01 | 5.000E+03 |
| CRB-14-1 | 1.300E-02 | PNL | BWR | 8/1/1985 | NA | | 8.979E+00 | 6.907E+02 |
| CRB-8-13471 | 1.300E-02 | SAIC | BWR | 09/15/1986 | 4.940E-02 | 3.800E+00 | 2.340E+02 | 1.800E+04 |
| CRB-3-1 | 1.200E-02 | SAIC | BWR | | 1.236E-01 | 1.030E+01 | NA | NA |
| CRB-3-2 | 1.200E-02 | SAIC | BWR | | 1.236E-01 | 1.030E+01 | NA | NA |
| CRB-11-11581 | 1.300E-02 | SAIC | BWR | 03/11/1985 | 3.900E-02 | 3.000E+00 | 7.540E-01 | 5.800E+01 |
| CRB-6-155 | 1.500E-02 | SAIC | BWR | <06/10/1985 | 4.800E-02 | 3.200E+00 | 7.500E+01 | 5.000E+03 |
| CRB-6-44 | 4.700E-02 | SAIC | BWR | <06/10/1985 | 1.504E-01 | 3.200E+00 | 2.914E+02 | 6.200E+03 |
| CRB-6-50 | 1.600E-02 | SAIC | BWR | <06/10/1985 | 5.120E-02 | 3.200E+00 | 2.080E+02 | 1.300E+04 |
| CRB-5-112 | 1.700E-02 | SAIC | BWR | <06/10/1985 | 5.440E-02 | 3.200E+00 | 1.700E+02 | 1.000E+04 |
| CRB-5-174 | 1.600E-02 | SAIC | BWR | <06/10/1985 | 5.120E-02 | 3.200E+00 | 1.760E+02 | 1.100E+04 |
| CRB-5-59 | 1.600E-02 | SAIC | BWR | <06/10/1985 | 5.120E-02 | 3.200E+00 | 1.216E+02 | 7.600E+03 |
| CRB-11-11580 | 1.300E-02 | SAIC | BWR | 03/11/1985 | 4.194E-02 | 3.226E+00 | 3.900E+02 | 3.000E+04 |
| CRB-9-105 | 1.800E-02 | TPT | BWR | 03/1984 | 4.004E-01 | | 1.700E+02 | 9.444E+03 |
| CRB-9-108 | 1.800E-02 | TPT | BWR | 03/1984 | 4.004E-01 | | 5.250E+02 | 2.917E+04 |
| CRB-9-32 | 1.800E-02 | TPT | BWR | 03/1984 | 4.004E-01 | | 2.220E+03 | 1.233E+05 |
| CRB-9-68 | 1.800E-02 | TPT | BWR | 03/1984 | 4.004E-01 | | 4.800E+02 | 2.667E+04 |
| CRB-9-71 | 1.810E-02 | WMG/RADCOR | BWR | 07/11/85 | 3.770E-01 | | 1.309E+03 | 7.230E+04 |
| CRB-9-75 | 1.810E-02 | WMG/RADCOR | BWR | 07/16/85 | 3.770E-01 | | 3.620E+02 | 2.000E+04 |
| CRB-9-7 | 1.810E-02 | WMG/RADCOR | BWR | 07/11/85 | 3.770E-01 | | 1.108E+03 | 6.120E+04 |
| CRB-9-92 | 1.800E-02 | TPT | BWR | 03/1984 | 4.004E-01 | | 9.300E+02 | 5.167E+04 |
| FUEL CHAN.-4-13095 | 6.100E-03 | SAIC | BWR | 05/06/1986 | 7.930E-02 | 1.300E+01 | 8.357E-02 | 1.370E+01 |
| FUEL CHANNEL-3-1# | 6.100E-03 | SAIC | BWR | | 6.100E-03 | 1.000E+00 | NA | NA |
| FUEL CHANNEL-5-1# | 6.100E-03 | SAIC | BWR | | 8.601E-02 | 1.410E+01 | NA | NA |
| FUEL CHANNEL-5-2# | 6.100E-03 | SAIC | BWR | | 8.601E-02 | 1.410E+01 | NA | NA |
| FUEL CHANNEL-5-3# | 6.100E-03 | SAIC | BWR | | 8.601E-02 | 1.410E+01 | NA | NA |
| FUEL CHANNEL-10-3# | 6.100E-03 | SAIC | BWR | | 7.930E-02 | 1.300E+01 | NA | NA |
| IRM-8-13472 | 3.210E-04 | SAIC | BWR | 09/15/1986 | 1.605E-02 | 5.000E+01 | 2.087E+02 | 6.500E+05 |
| IRM-14-1# | 3.210E-04 | SAIC | BWR | | 1.589E-02 | 4.950E+01 | NA | NA |
| LPRM-4-13031 | 1.990E-04 | SAIC | BWR | 05/6/1986 | 8.975E-03 | 4.510E+01 | 1.373E+02 | 6.900E+05 |
| LPRM-4-13032 | 2.070E-04 | SAIC | BWR | 05/6/1986 | 9.336E-03 | 4.510E+01 | 4.140E+01 | 2.000E+05 |
| LPRM-8-13470 | 1.970E-04 | SAIC | BWR | 09/15/1986 | 1.361E-02 | 6.910E+01 | 8.471E+01 | 4.300E+05 |
| LPRM-14-1 | 1.600E-04 | PNL | BWR | 8/1/1985 | 5.440E-04 | 3.400E+00 | 7.792E+00 | 4.870E+04 |
| LPRM-1-1 | 1.600E-04 | SAIC | BWR | <04/22/1985 | 1.333E-02 | 8.333E+01 | 3.680E+01 | 2.300E+05 |
| LPRM-11-11582 | 5.000E-05 | SAIC | BWR | 03/11/1985 | 2.500E-03 | 5.000E+01 | 1.400E+01 | 2.800E+05 |
| LPRM-14-2 | 1.600E-04 | PNL | BWR | 8/1/1985 | 5.424E-04 | 3.390E+00 | 9.040E+00 | 5.650E+04 |
| LPRM-1-2 | 1.400E-04 | SAIC | BWR | <04/22/1985 | 1.167E-02 | 8.333E+01 | 1.680E+01 | 1.200E+05 |
| LPRM-14-3 | 1.600E-04 | PNL | BWR | 8/1/1985 | 5.408E-04 | 3.380E+00 | 1.411E+01 | 8.820E+04 |
| LPRM-2-1 | 1.600E-04 | SAIC | BWR | | 7.280E-03 | 4.550E+01 | NA | NA |
| LPRM-3-1 | 1.600E-04 | SAIC | BWR | | 6.960E-03 | 4.350E+01 | NA | NA |
| LPRM-6-1 | 1.400E-04 | SAIC | BWR | <06/10/1985 | 7.778E-03 | 5.556E+01 | 3.920E+01 | 2.800E+05 |
| LPRM-5-1 | 1.400E-04 | SAIC | BWR | <06/10/1985 | 7.784E-03 | 5.560E+01 | 7.280E+01 | 5.200E+05 |
| LPRM-6-2 | 1.300E-04 | SAIC | BWR | <06/10/1985 | 7.222E-03 | 5.556E+01 | 7.410E+01 | 5.700E+05 |
| LPRM-5-2 | 1.400E-04 | SAIC | BWR | <06/10/1985 | 7.784E-03 | 5.560E+01 | 4.760E+01 | 3.400E+05 |

## Table 4.1 Hardware Characteristic Summary (continued)

| HARDWARE KIND | HARDWARE VOLUME (M**3) | METHOD | PLANT TYPE | CHARCTER-IZATION DATE | DOSE CURIE CO-60 CONV. (CI/R/HR) | DOSE CURIE CO-60 CONV. (CI/M^^3/R/HR) | Co-60 (Ci) | (CI/M^^3) CO-60 |
|---|---|---|---|---|---|---|---|---|
| LPRM-10-1# | 1.400E-04 | SAIC | BWR | | 9.338E-03 | 6.670E+01 | NA | NA |
| LPRM-12-2 | 5.000E-05 | SAIC | BWR | | 3.615E-03 | 7.230E+01 | NA | NA |
| LPRM-14-1 | 1.600E-04 | SAIC | BWR | | 7.216E-03 | 4.510E+01 | NA | NA |
| LPRM-14-2 | 1.600E-04 | SAIC | BWR | | 7.216E-03 | 4.510E+01 | NA | NA |
| LPRM-4-13033 | 1.970E-04 | SAIC | BWR | 05/6/1986 | 8.885E-03 | 4.510E+01 | 6.107E+01 | 3.100E+05 |
| | | | | | | | | |
| POISON CUR.-11-11579 | 2.100E-02 | SAIC | BWR | 03/08/1985 | 6.174E-01 | 2.940E+01 | 1.512E+03 | 7.200E+04 |
| POISON CUR.-8-1# | 2.100E-02 | SAIC | BWR | | 6.174E-01 | 2.940E+01 | NA | NA |
| POISON CUR.-8-2# | 2.100E-02 | SAIC | BWR | | 6.174E-01 | 2.940E+01 | NA | NA |
| | | | | | | | | |
| BPRA-13-A9P21 | 2.600E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.560E-01 | | 4.082E+01 | 1.570E+04 |
| BPRA-13-12P27 | 3.962E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.560E-01 | | 4.120E+01 | 1.040E+04 |
| BPRA-13-16P73 | 4.953E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.560E-01 | | 4.135E+01 | 8.350E+03 |
| BPRA-13-16P78 | 4.953E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.560E-01 | | 4.135E+01 | 8.350E+03 |
| BPRA-15-2 | 3.879E-03 | WMG/RADCOR | PWR | 06/24/86 | 2.560E-01 | | 1.478E+02 | 3.810E+04 |
| BPRA-7-2 | 4.890E-03 | WMG/RADCOR | PWR | 11/04/85 | 2.560E-01 | | 2.445E+01 | 5.000E+03 |
| BPRA-13-20P56 | 5.943E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.560E-01 | | 4.136E+01 | 6.960E+03 |
| BPRA-13-20P70 | 5.943E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.560E-01 | | 4.136E+01 | 6.960E+03 |
| BPRA-13-B10P21 | 3.311E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.560E-01 | | 4.139E+01 | 1.250E+04 |
| | | | | | | | 0.000E+00 | |
| GUIDE TUBE-15-1 | 9.911E-04 | WMG/RADCOR | PWR | 06/30/86 | 1.850E-01 | | 6.868E+01 | 6.930E+04 |
| GUIDE TUBES-7-003 | 8.490E-05 | WMG/RADCOR | PWR | 02/18/86 | 2.180E-01 | | 2.988E+01 | 3.520E+05 |
| | | | | | | | 0.000E+00 | |
| PRIM SCR-13-APS2 | 4.104E-03 | WMG/RADCOR | PWR | 10/10/86 | 4.500E-02 | | 4.145E+01 | 1.010E+04 |
| PSA-15-1 | 4.106E-03 | WMG/RADCOR | PWR | 06/24/86 | 4.500E-02 | | 1.080E+02 | 2.630E+04 |
| SEC SCR-13-SS4 | 4.104E-03 | WMG/RADCOR | PWR | 10/10/86 | 4.500E-02 | | 4.145E+01 | 1.010E+04 |
| | | | | | | | 0.000E+00 | |
| RCCA-13-R123 | 7.075E-03 | WMG/RADCOR | PWR | 10/10/86 | 2.880E-01 | | 4.556E+02 | 6.440E+04 |
| RCCA-15-1 | 8.530E-02 | WMG/RADCOR | PWR | 11/04/85 | 2.880E-01 | | 9.810E+00 | 1.150E+02 |
| RCCA-7-1 | 7.079E-03 | WMG/RADCOR | PWR | 06/24/86 | 2.880E-01 | | 2.669E+01 | 3.770E+03 |
| | | | | | | | 0.000E+00 | |
| THIMBLE PLUG-7-10 | 6.790E-04 | WMG/RADCOR | PWR | 11/04/85 | 2.400E-02 | | 1.548E+01 | 2.280E+04 |
| THIMBLE PLUG-7-12 | 6.790E-04 | WMG/RADCOR | PWR | 11/09/85 | 2.400E-02 | | 4.828E+01 | 7.110E+04 |
| THIMBLE PLUG-13-215 | 6.226E-04 | WMG/RADCOR | PWR | 10/10/86 | 2.400E-02 | | 4.950E+00 | 7.950E+03 |
| THIMBLE PLUGS-15-6 | 6.227E-04 | WMG/RADCOR | PWR | 06/24/86 | 2.400E-02 | | 1.930E+01 | 3.100E+04 |

# BASED ON A ANOTHER PIECES VOLUME

2. Most of the data applies to BWR components,

3. $^{60}$Co concentrations for PWR components are generally lower than those for BWR components,

4. The displacement volumes used for the same types of components vary by up to 70 percent between plants and methods,

5. The $^{60}$Co dose-to-curie conversion factors used with the direct sample method for the same type of component vary significantly by plant.

Although sparse, these data should enable individual readers to generally assess the characteristics of their plant specific hardware.

## 4.2 Scaling Factors

Table 4.2 shows reported scaling factors for the same components identified in Table 4.1. Scaling factors are listed for Part 61 radionuclides which affect classification, plus $^{55}$Fe. The information obtained from activation analysis characterizations also shows irradiation history as available.

There are differences in the basis used to determine scaling factors. For the activation analysis method all values are calculated. The direct sample method measures $^{63}$Ni, $^{14}$C, $^{99}$Tc, $^{94}$Nb and $^{55}$Fe relative to $^{60}$Co with the reported values for $^{94}$Nb generally being LLD values. The reported $^{59}$Ni values are based on applying a generic scaling factor to the $^{63}$Ni value.

The ranges of scaling factors shown in Table 4.2 vary, but the information to make comparisons by component and method is presented. There are, however, some method dependent differences that warrant discussion. These differences, which address the scaling factors for $^{63}$Ni, $^{59}$Ni, and $^{94}$Nb, are presented below.

### 4.2.1 $^{59}$Ni and $^{63}$Ni Scaling

Figure 4.1 shows the relationship between $^{59}$Ni/$^{63}$Ni for CRB's for both the direct measurement and the activation analysis methods. The same relationship is shown on Figure 4.2 for stainless tubing components, BWR instrument strings, and PWR guide tubes. The Figures show a clear difference between the $^{59}$Ni/$^{63}$Ni relationship for the two methods. The direct sample method relies on a $^{59}$Ni scaling factor higher than that determined by the activation analysis methods. This discrepancy is analyzed in Appendix B.

# Table 4.2 Part 61 Nuclide Scaling Factors

| HARDWARE KIND | METHOD | PLANT TYPE | EFPY (MWD/MTU) | REMOVAL DATE | C-14/CO-60 | FE-55/CO-60 | NI-59/CO-60 | NI-63/CO-60 | NB-94/CO-60 | TC-99/CO-60 |
|---|---|---|---|---|---|---|---|---|---|---|
| CRB-12-1# | SAIC | BWR | | | 5.600E-06 | 1.100E+00 | 1.200E-03 | 6.000E-02 | 1.500E-07 | NA |
| CRB-4-13035 | SAIC | BWR | | | 3.784E-06 | 1.844E+00 | ND | 6.048E-02 | 1.617E-07 | ND |
| CRB-1-1 | SAIC | BWR | | | 2.097E-07 | 2.649E-01 | 2.208E-03 | 1.004E-01 | 2.649E-06 | 1.656E-06 |
| CRB-14-1 | PNL | BWR | | | 8.832E-05 | 9.191E-01 | 5.791E-04 | 7.239E-02 | 2.027E-06 | 2.027E-07 |
| CRB-8-13471 | SAIC | BWR | | | 4.835E-06 | 6.209E-01 | ND | 1.006E-01 | 1.329E-06 | ND |
| CRB-3-1 | SAIC | BWR | | | 1.200E-06 | 2.500E+00 | NA | 1.100E-01 | 5.700E-07 | NA |
| CRB-3-2 | SAIC | BWR | | | 5.300E-07 | 1.400E+00 | NA | 6.600E-02 | 3.200E-07 | NA |
| CRB-11-11581 | SAIC | BWR | | | 9.754E-04 | 7.631E-01 | 3.361E-03 | 8.361E-02 | 3.770E-06 | ND |
| CRB-6-155 | SAIC | BWR | | | 4.200E-06 | 8.800E-01 | 4.600E-03 | 4.000E-02 | 9.400E-07 | NA |
| CRB-6-44 | SAIC | BWR | | | 4.194E-06 | 8.226E-01 | 4.516E-03 | 3.871E-02 | 9.032E-07 | NA |
| CRB-6-50 | SAIC | BWR | | | 4.154E-06 | 8.462E-01 | 4.462E-03 | 4.000E-02 | 9.231E-07 | NA |
| CRB-5-112 | SAIC | BWR | | | 8.333E-06 | 7.750E-01 | 2.083E-03 | 1.167E-01 | 4.750E-06 | 7.750E-09 |
| CRB-5-174 | SAIC | BWR | | | 8.750E-06 | 6.875E-01 | 2.063E-03 | 1.188E-01 | 4.688E-06 | 6.250E-09 |
| CRB-5-59 | SAIC | BWR | | | 9.167E-06 | 7.917E-01 | 2.083E-03 | 1.167E-01 | 4.833E-06 | 5.917E-09 |
| CRB-11-11580 | SAIC | BWR | | | 1.074E-05 | 8.807E-01 | 4.428E-03 | 9.619E-02 | 3.075E-07 | ND |
| CRB-9-105 | TPT | BWR | | | 3.353E-04 | 8.235E-01 | 2.294E-03 | 2.706E-01 | 4.600E-06 | 5.547E-07 |
| CRB-9-108 | TPT | BWR | | | 1.714E-04 | 1.695E+00 | 1.105E-03 | 1.371E-01 | 3.371E-06 | 2.667E-07 |
| CRB-9-32 | TPT | BWR | | | 9.459E-05 | 9.865E-01 | 6.351E-04 | 7.883E-02 | 1.288E-06 | 1.550E-07 |
| CRB-9-68 | TPT | BWR | | | 1.667E-04 | 1.688E+00 | 6.875E-04 | 1.375E-01 | 2.208E-06 | 2.667E-07 |
| CRB-9-71 | WMG/RADCOR | BWR | 8.270E+00 | 02/12/83 | 1.397E-04 | 2.006E+00 | 9.696E-04 | 1.122E-01 | 7.939E-07 | 2.209E-06 |
| CRB-9-75 | WMG/RADCOR | BWR | 4.200E+00 | 12/27/75 | 2.815E-04 | 6.209E-01 | 1.960E-03 | 2.175E-01 | 1.329E-06 | 2.010E-06 |
| CRB-9-7 | WMG/RADCOR | BWR | 6.950E+00 | 01/05/80 | 1.961E-04 | 1.371E+00 | 1.361E-03 | 1.542E-01 | 1.109E-06 | 3.154E-06 |
| CRB-9-92 | TPT | BWR | | | 1.613E-04 | 1.710E+00 | 1.097E-03 | 1.398E-01 | 2.237E-06 | 2.688E-07 |
| FUEL CHAN.-4-13095 | SAIC | BWR | | | 1.606E-05 | 1.102E+00 | ND | 1.066E-01 | 2.847E-05 | ND |
| FUEL CHANNEL-3-1# | SAIC | BWR | | | 1.900E+05 | 3.100E+00 | NA | 5.600E-02 | 3.000E-06 | NA |
| FUEL CHANNEL-5-1# | SAIC | BWR | | | 4.900E-07 | 1.400E-01 | NA | 7.000E-03 | 3.000E-06 | NA |
| FUEL CHANNEL-5-2# | SAIC | BWR | | | 1.800E-06 | 5.100E-01 | NA | 2.300E-02 | 3.500E-06 | NA |
| FUEL CHANNEL-5-3# | SAIC | BWR | | | 1.500E-06 | 3.700E-01 | NA | 2.100E-02 | 3.600E-06 | NA |
| FUEL CHANNEL-10-3# | SAIC | BWR | | | 2.900E-04 | 2.600E-01 | 1.000E-03 | 3.600E-02 | 2.000E-04 | NA |
| IRM-8-13472 | SAIC | BWR | | | 2.235E-05 | 3.962E+00 | ND | 2.407E-01 | 9.317E-06 | ND |
| IRM-14-1# | SAIC | BWR | | | 6.600E-07 | 7.300E-01 | 9.100E-04 | 4.700E-02 | 5.700E-08 | NA |
| LPRM-4-13031 | SAIC | BWR | | | 3.302E-06 | 8.983E-01 | ND | 3.700E-02 | 1.572E-07 | ND |
| LPRM-4-13032 | SAIC | BWR | | | 4.816E-06 | 1.575E+00 | ND | 4.881E-02 | 3.688E-07 | ND |
| LPRM-8-13470 | SAIC | BWR | | | 2.303E-05 | 3.743E+00 | ND | 5.487E-02 | 4.408E-07 | ND |
| LPRM-14-1 | PNL | BWR | | | 5.585E-04 | 7.823E-01 | 3.265E-03 | 4.189E-01 | 4.559E-06 | 1.170E-06 |
| LPRM-1-1 | SAIC | BWR | | | 1.696E-06 | 1.261E+00 | 1.609E-03 | 6.522E-02 | 1.217E-06 | 4.130E-07 |
| LPRM-11-11582 | SAIC | BWR | | | 1.196E-05 | 1.134E+00 | 1.375E-03 | 6.813E-02 | 1.696E-07 | ND |
| LPRM-14-2 | PNL | BWR | | | 5.575E-04 | 7.823E-01 | 3.257E-03 | 4.177E-01 | 4.566E-06 | 1.170E-06 |
| LPRM-1-2 | SAIC | BWR | | | 1.691E-06 | 1.250E+00 | 1.618E-03 | 6.618E-02 | 1.176E-07 | 6.176E-07 |
| LPRM-14-3 | PNL | BWR | | | 5.578E-04 | 7.823E-01 | 3.265E-03 | 4.184E-01 | 4.558E-06 | 1.168E-06 |
| LPRM-2-1 | SAIC | BWR | | | 2.800E-05 | 1.500E+00 | NA | 3.800E-01 | 2.000E-07 | NA |
| LPRM-3-1 | SAIC | BWR | | | 8.900E-07 | 6.700E-01 | NA | 5.200E-02 | 1.000E-07 | NA |
| LPRM-6-1 | SAIC | BWR | | | 8.750E-06 | 1.500E+00 | 7.917E-04 | 3.750E-02 | 4.167E-07 | 2.500E-09 |
| LPRM-5-1 | SAIC | BWR | | | 5.614E-06 | 9.649E-01 | 1.140E-03 | 5.965E-02 | 4.035E-07 | NA |
| LPRM-6-2 | SAIC | BWR | | | 8.788E-06 | 1.500E+00 | 7.879E-04 | 3.636E-02 | 5.909E-07 | 2.879E-09 |
| LPRM-5-2 | SAIC | BWR | | | 5.357E-06 | 1.000E+00 | 1.107E-03 | 5.714E-02 | 3.929E-07 | NA |

## Table 4.2 Part 61 Nuclide Scaling Factors (continued)

| HARDWARE KIND | METHOD | PLANT TYPE | EFPY (MWD/MTU) | REMOVAL DATE | C-14/CO-60 | FE-55/CO-60 | NI-59/CO-60 | NI-63/CO-60 | NB-94/CO-60 | TC-99/CO-60 |
|---|---|---|---|---|---|---|---|---|---|---|
| LPRM-10-1# | SAIC | BWR | | | NA | 2.600E-01 | 7.800E-04 | 4.700E-02 | 6.000E-07 | NA |
| LPRM-12-2 | SAIC | BWR | | | 2.100E-06 | 1.000E+00 | 8.400E-04 | 4.300E-02 | 6.200E-07 | NA |
| LPRM-14-1 | SAIC | BWR | | | 3.400E-06 | 7.300E-01 | 7.400E-04 | 3.800E-02 | 1.900E-07 | NA |
| LPRM-14-2 | SAIC | BWR | | | 4.500E-06 | 6.400E-01 | 5.900E-04 | 3.000E-02 | 1.800E-07 | NA |
| LPRM-4-13033 | SAIC | BWR | | | 8.443E-06 | 8.017E-01 | ND | 3.330E-02 | 8.696E-08 | ND |
| POISON CUR.-11-115 | SAIC | BWR | | | 2.815E-06 | 4.558E-01 | 3.646E-03 | 1.810E-01 | 5.761E-06 | ND |
| POISON CUR.-8-1# | SAIC | BWR | | | 8.800E-06 | 1.000E-01 | NA | 5.100E-02 | 8.000E-07 | NA |
| POISON CUR.-8-2# | SAIC | BWR | | | 4.600E-06 | 2.500E-01 | NA | 2.800E-02 | 5.000E-07 | NA |
| BPRA-13-A9P21 | WMG/RADCOR | PWR | 6.400E-01 | 01/21/83 | 9.745E-05 | 2.420E+00 | 7.580E-04 | 8.854E-02 | 9.427E-07 | 1.904E-07 |
| BPRA-13-12P27 | WMG/RADCOR | PWR | 1.200E+00 | 01/21/83 | 1.019E-04 | 2.356E+00 | 7.885E-04 | 9.221E-02 | 9.808E-07 | 1.962E-07 |
| BPRA-13-16P73 | WMG/RADCOR | PWR | 1.410E+00 | 01/21/83 | 1.028E-04 | 2.323E-01 | 7.976E-04 | 9.305E-02 | 9.880E-07 | 2.000E-07 |
| BPRA-13-16P78 | WMG/RADCOR | PWR | 1.820E+00 | 10/04/84 | 8.407E-05 | 2.802E+00 | 6.527E-04 | 7.713E-02 | 8.120E-07 | 1.629E-07 |
| BPRA-15-2 | WMG/RADCOR | PWR | 1.070E+00 | 08/16/84 | 1.034E-04 | 2.913E+00 | 7.927E-04 | 9.344E-02 | 1.205E-06 | 1.942E-07 |
| BPRA-7-2 | WMG/RADCOR | PWR | 1.110E+00 | 10/31/75 | 2.620E-04 | 1.078E+00 | 1.906E-03 | 2.140E-01 | 2.880E-06 | 3.340E-07 |
| BPRA-13-20P56 | WMG/RADCOR | PWR | 1.720E+00 | 10/04/84 | 8.362E-05 | 2.830E+00 | 6.494E-04 | 7.658E-02 | 8.046E-07 | 2.026E-07 |
| BPRA-13-20P70 | WMG/RADCOR | PWR | 1.370E+00 | 01/21/83 | 1.024E-04 | 2.328E+00 | 7.945E-04 | 9.282E-02 | 9.856E-07 | 1.997E-07 |
| BPRA-13-B10P21 | WMG/RADCOR | PWR | 9.000E-01 | 01/21/83 | 9.920E-05 | 2.376E+00 | 7.728E-04 | 9.040E-02 | 9.600E-07 | 1.928E-07 |
| GUIDE TUBE-15-1 | WMG/RADCOR | PWR | 1.070E+00 | 10/25/82 | 1.147E-04 | 2.612E+00 | 8.167E-04 | 9.610E-02 | 1.234E-06 | 1.299E-07 |
| GUIDE TUBES-7-003 | WMG/RADCOR | PWR | 2.180E+00 | 05/04/74 | 3.438E-04 | 7.756E-01 | 2.497E-03 | 2.756E-01 | 3.295E-06 | 4.006E-07 |
| PRIM SCR-13-APS2 | WMG/RADCOR | PWR | 1.260E+00 | 01/21/83 | 1.010E-04 | 2.347E+00 | 7.931E-04 | 9.257E-02 | 1.000E-06 | 1.980E-07 |
| PSA-15-1 | WMG/RADCOR | PWR | 1.070E+00 | 08/16/84 | 1.027E-04 | 2.909E+00 | 8.023E-04 | 9.544E-02 | 1.259E-06 | 2.046E-07 |
| SEC SCR-13-SS4 | WMG/RADCOR | PWR | 1.810E+00 | 10/04/84 | 8.347E-05 | 2.822E+00 | 6.554E-04 | 7.743E-02 | 8.297E-07 | 1.634E-07 |
| RCCA-13-R123 | WMG/RADCOR | PWR | 1.340E+00 | 01/21/83 | 1.014E-05 | 2.391E-01 | 7.904E-05 | 9.193E-03 | 9.348E-08 | 1.724E-08 |
| RCCA-15-1 | WMG/RADCOR | PWR | 8.200E-01 | 10/31/75 | 2.991E-04 | 9.391E-01 | 2.209E-03 | 2.443E-01 | 4.409E-06 | 4.409E-07 |
| RCCA-7-1 | WMG/RADCOR | PWR | 8.200E-01 | 10/25/84 | 8.249E-05 | 3.422E+00 | 1.019E-03 | 1.215E-01 | 1.753E-06 | 6.101E-07 |
| THIMBLE PLUG-7-10 | WMG/RADCOR | PWR | 6.610E+00 | 08/08/80 | 1.886E-04 | 1.522E+00 | 2.136E-03 | 2.425E-01 | 3.943E-06 | 1.189E-06 |
| THIMBLE PLUG-7-12 | WMG/RADCOR | PWR | 7.450E+00 | 02/26/82 | 1.617E-04 | 1.828E+00 | 1.828E-03 | 2.096E-01 | 3.376E-06 | 1.018E-06 |
| THIMBLE PLUG-13-21 | WMG/RADCOR | PWR | 1.830E+00 | 10/04/84 | 7.975E-05 | 2.704E+00 | 1.014E-03 | 1.194E-01 | 1.472E-06 | 5.673E-07 |
| THIMBLE PLUGS-15-8 | WMG/RADCOR | PWR | 1.070E+00 | 10/25/82 | 1.081E-04 | 2.497E-01 | 1.419E-03 | 1.668E-01 | 2.590E-07 | 9.226E-08 |

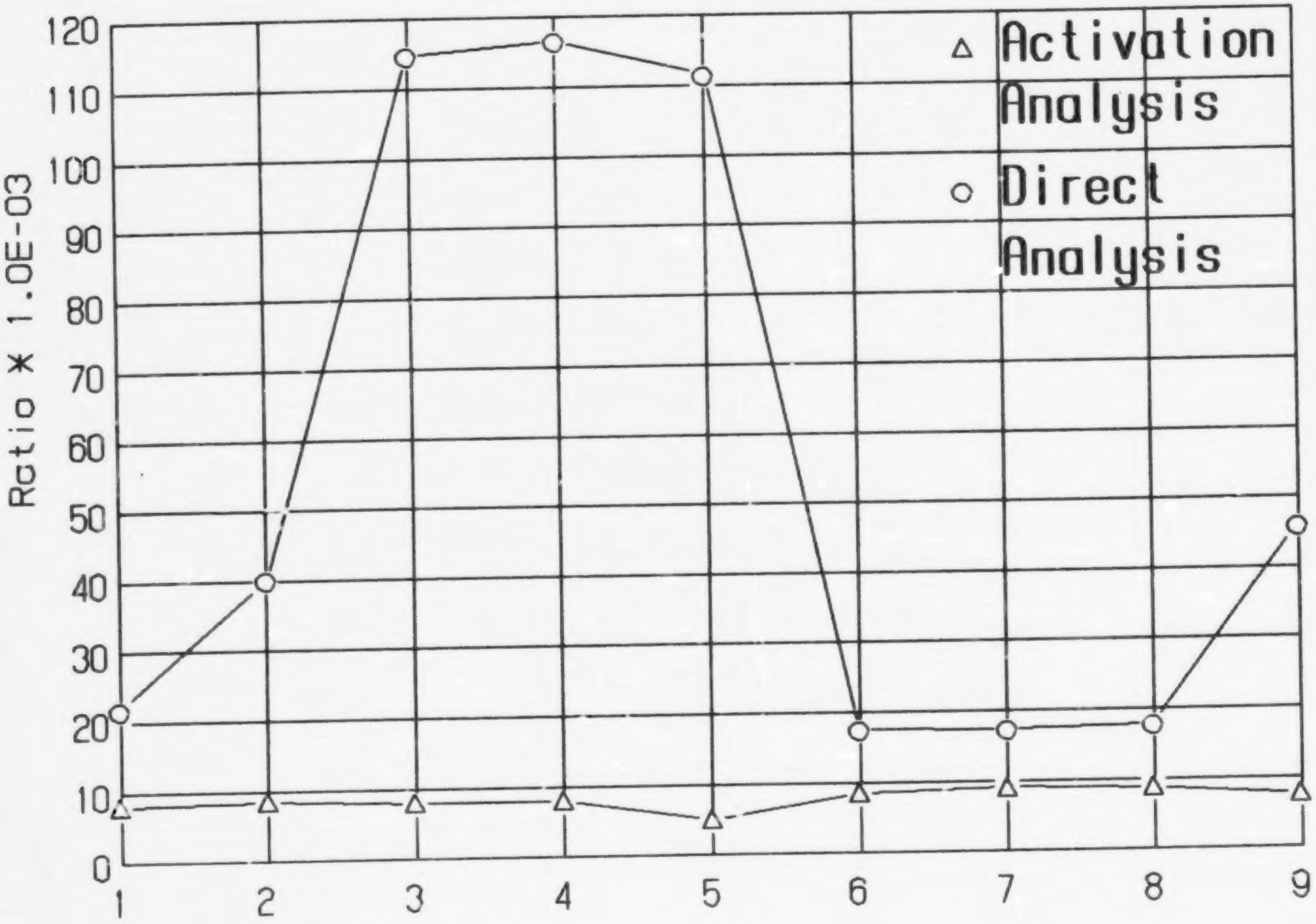

**Figure 4.1** $^{59}Ni/^{63}Ni$ For Control Rod Blades

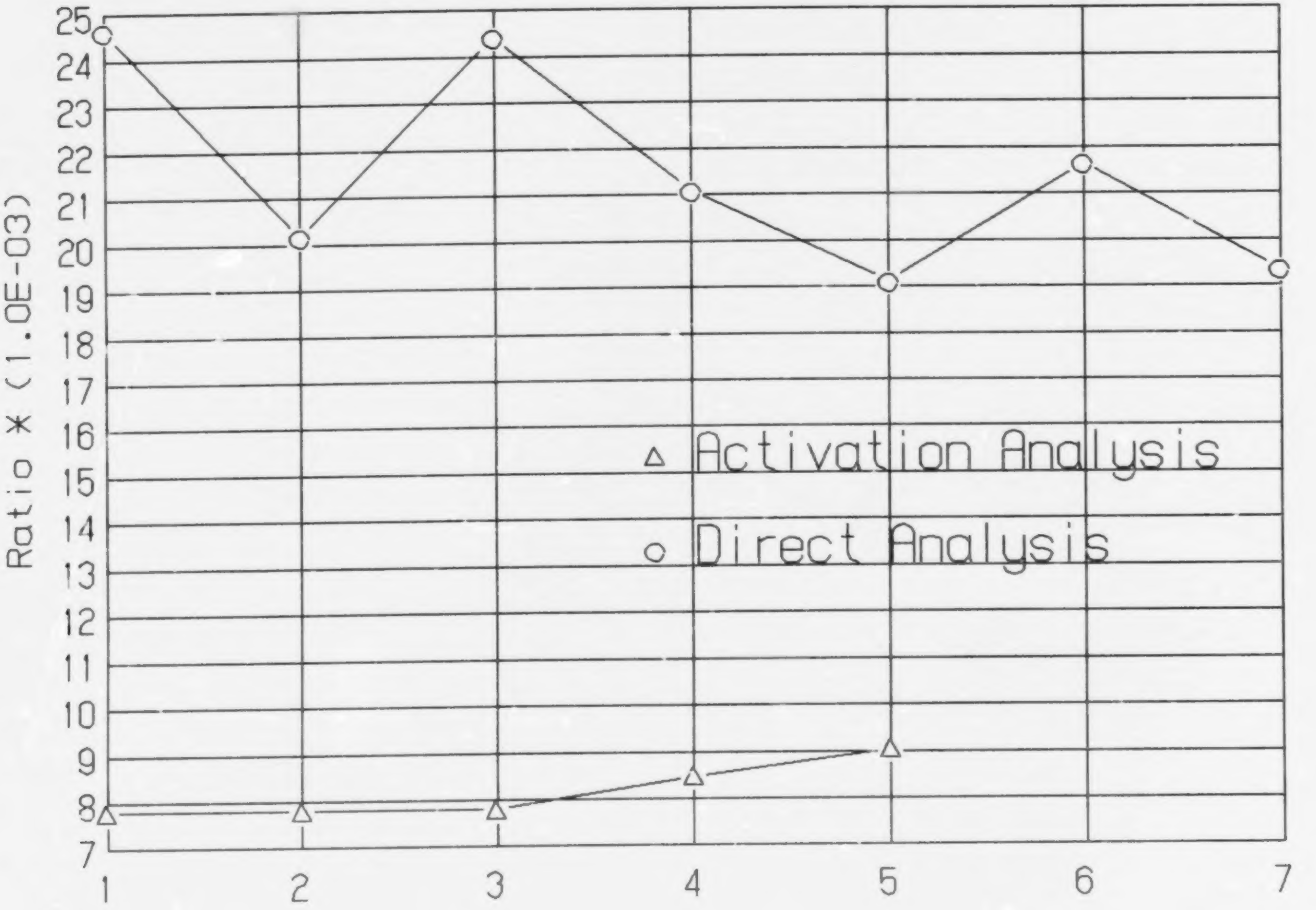

**Figure 4.2** $^{59}Ni/^{63}Ni$ For LPRM's

### 4.2.2 $^{94}$Nb Scaling Factors

The scaling factors reported for the direct sample method are generally based on lower limit of detection (LLD) values, while those reported for activation analysis methods are calculated. Figure 4.3 compares the $^{94}$Nb scaling factors for BWR CRB's, while Figure 4.4 shows a similar comparison for stainless steel tubing, i.e., LPRM's and guide tubes.

As shown on Figure 4.3, the scaling factors for CRB's for both methods vary over a range of 5E-7 to 5E-6 and are comparable. Since the direct sample method usually reports LLD values, all methods appear to be conservative when determining $^{94}$Nb.

Figure 4.4 indicates a consistent method dependent difference in the $^{94}$Nb scaling factors for LPRM's and guide tubes. The direct sample method results show scaling factors at or below 5E-7 while the activation analysis methods show values from 1E-6 to as high as 5E-6. Again, however, since the direct sample method usually reports LLD values, both methods appear to be conservative when determining $^{94}$Nb.

### 4.3 Basis for Part 61 Classification

Classification of hardware generally depends on the concentrations of $^{59}$Ni, $^{14}$C, $^{99}$Tc and $^{94}$Nb relative to Table 1 limits or the concentration of $^{63}$Ni relative to its Table 2 limit. Table 4.3 divides the reported concentrations of these radionuclides in the data base by the relevant Part 61 limits, and presents these values as fractions of the limits. It also shows the sum of the fractions relative to unity for both Tables and indicates the controlling Table for classification. As shown, where the activation analysis method is applied, $^{63}$Ni concentrations almost always determine Part 61 classification. Conversely, where the direct sample method is used, two-thirds of the time, the sum of the fractions for Table 1 radionuclides governs classification.

The Table also illustrates the relative importance of the Part 61 radionuclides in hardware classification.

Classification of component types relative to Part 61 Class C limits can also be discerned from the sum of fractions information. Control rod blades cover the entire gamut of fractions and in some cases slightly exceed Class C limits on an individual component basis. The stainless steel tube-like components - BWR instrument strings, and PWR guide tubes - generally fall at the upper range of fractions, and in many cases are well above Class C concentration limits on an individual component basis. On the other hand, PWR components, except thimble plug assemblies, fall in the lower range of Class C limit fractions. Thimble plug assemblies cover the full range of Class C fractions, and on an individual component basis can exceed Class C limits.

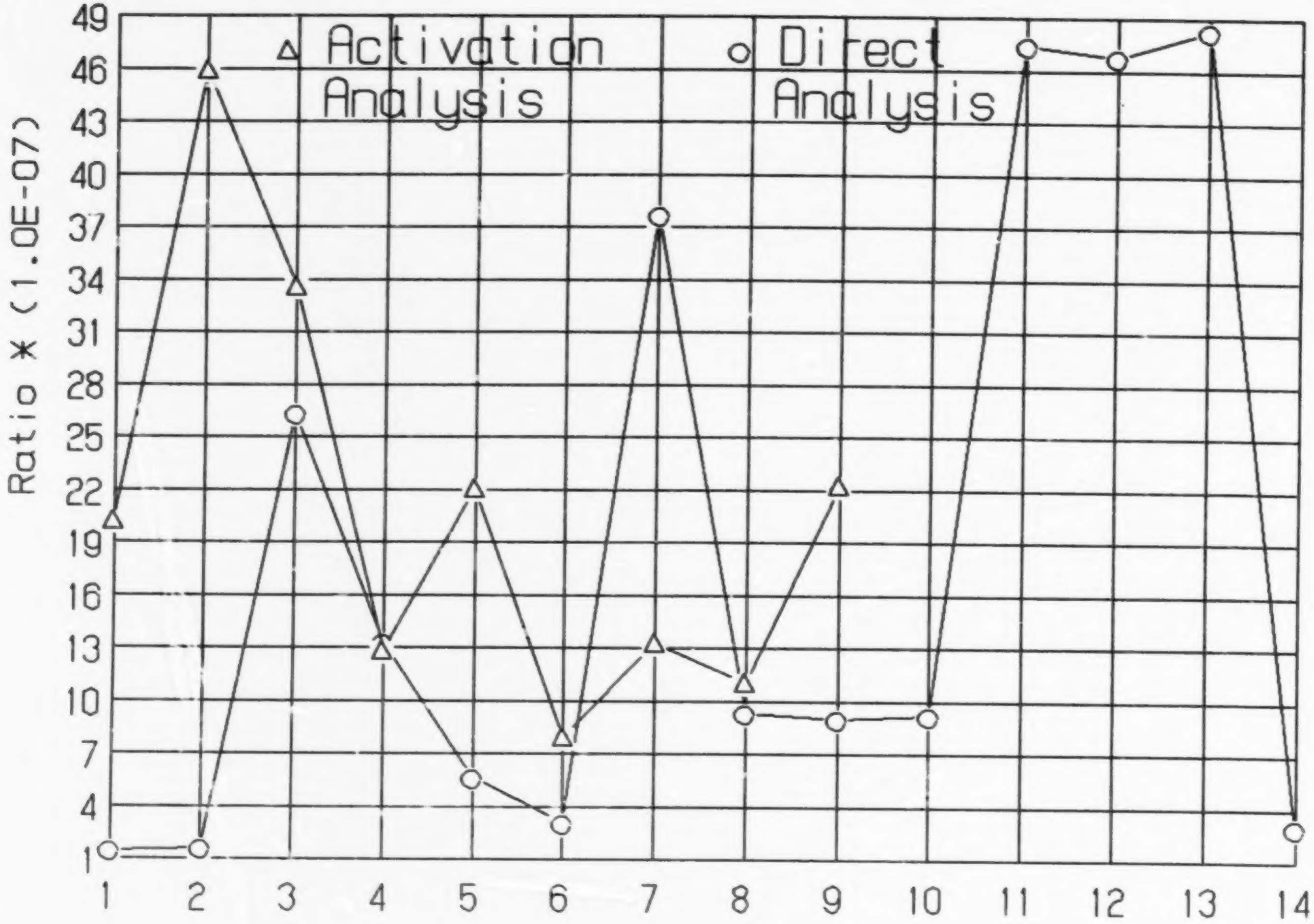

__Figure 4.3__ $^{94}$Nb Scaling Factors For Control Rod Blades

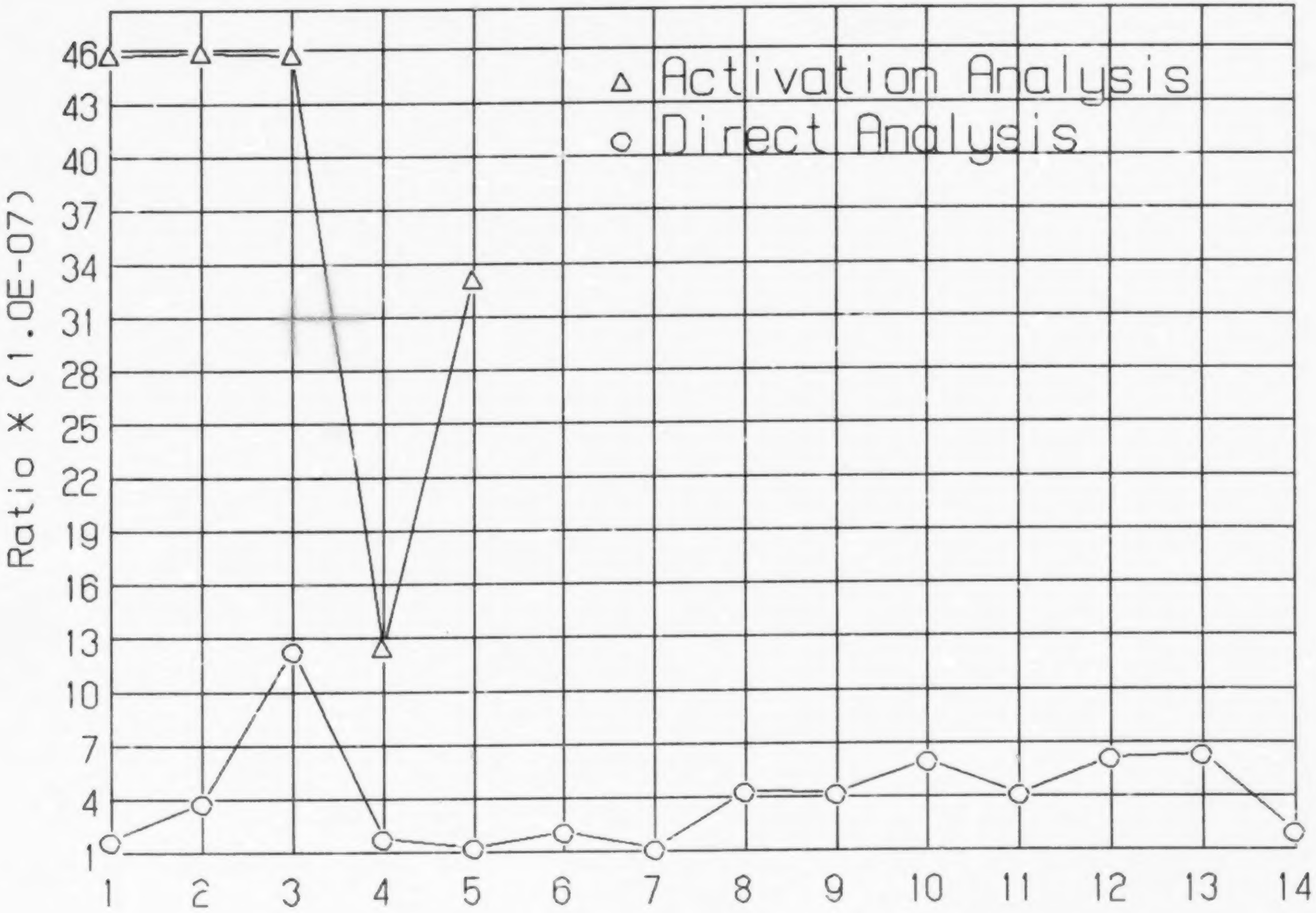

Figure 4.4 $^{94}$Nb Scaling Factors For LPRM's

# Table 4.3 Part 61 Classification Summary

| HARDWARE KIND | METHOD | PLANT TYPE | C-14/ LIMIT | NI-59 / LIMIT | NB-94 / LIMIT | TC-99 / LIMIT | SUM OF THE TABLE 1 FRACS | NI-63 / LIMIT | SUM OF THE TABLE 2 FRACS | DETRMN. TABLE |
|---|---|---|---|---|---|---|---|---|---|---|
| CRB-4-13035 | SAIC | BWR | 0.00001 | ND | 0.00013 | ND | 0.00014 | 0.00144 | 0.00144 | 2 |
| CRB-1-1 | SAIC | BWR | 0.00002 | 0.09091 | 0.12000 | 0.00500 | 0.21593 | 0.13000 | 0.13000 | 1 |
| CRB-14-1 | PNL | BWR | 0.00076 | 0.00182 | 0.00700 | 0.00005 | 0.00963 | 0.00714 | 0.00714 | 1 |
| CRB-8-13471 | SAIC | BWR | 0.00001 | ND | 0.00105 | ND | 0.00106 | 0.00227 | 0.00227 | 2 |
| CRB-11-11581 | SAIC | BWR | 0.00015 | 0.00019 | 0.00023 | ND | 0.00057 | 0.00015 | 0.00015 | 1 |
| CRB-6-155 | SAIC | BWR | 0.00026 | 0.10455 | 0.02350 | NA | 0.12831 | 0.02857 | 0.02857 | 1 |
| CRB-6-44 | SAIC | BWR | 0.00033 | 0.12727 | 0.02800 | NA | 0.15560 | 0.03429 | 0.03429 | 1 |
| CRB-6-50 | SAIC | BWR | 0.00068 | 0.26364 | 0.06000 | NA | 0.32431 | 0.07429 | 0.07429 | 1 |
| CRB-5-112 | SAIC | BWR | 0.00125 | 0.11364 | 0.28500 | 0.00003 | 0.39992 | 0.20000 | 0.20000 | 1 |
| CRB-5-174 | SAIC | BWR | 0.00175 | 0.15000 | 0.37500 | 0.00003 | 0.52678 | 0.27143 | 0.27143 | 1 |
| CRB-5-59 | SAIC | BWR | 0.00138 | 0.11364 | 0.29000 | 0.00002 | 0.40504 | 0.20000 | 0.20000 | 1 |
| CRB-11-11580 | SAIC | BWR | 0.00109 | 0.16364 | 0.01250 | ND | 0.17723 | 0.11171 | 0.11171 | 1 |
| CRB-9-105 | TPT | BWR | 0.03958 | 0.09848 | 0.21722 | 0.00175 | 0.35704 | 0.36508 | 0.36508 | 2 |
| CRB-9-108 | TPT | BWR | 0.06250 | 0.14646 | 0.49167 | 0.00259 | 0.70322 | 0.57143 | 0.57143 | 1 |
| CRB-9-32 | TPT | BWR | 0.14583 | 0.35606 | 0.79444 | 0.00637 | 1.30271 | 1.38889 | 1.38889 | 2 |
| CRB-9-68 | TPT | BWR | 0.05556 | 0.08333 | 0.29444 | 0.00237 | 0.43570 | 0.52381 | 0.52381 | 2 |
| CRB-9-71 | WMG/RADCOR | BWR | 0.12625 | 0.31864 | 0.28700 | 0.05323 | 0.78512 | 1.15857 | 1.15857 | 2 |
| CRB-9-75 | WMG/RADCOR | BWR | 0.07038 | 0.17818 | 0.15950 | 0.01340 | 0.42146 | 0.62143 | 0.62143 | 2 |
| CRB-9-7 | WMG/RADCOR | BWR | 0.15000 | 0.37864 | 0.33950 | 0.06433 | 0.93247 | 1.34857 | 1.34857 | 2 |
| CRB-9-92 | TPT | BWR | 0.10417 | 0.25758 | 0.57778 | 0.00463 | 0.94415 | 1.03175 | 1.03175 | 2 |
| FUEL CHAN.-4-13 | SAIC | BWR | .00000 | ND | 0.00195 | ND | 0.00195 | 0.00021 | 0.00021 | 1 |
| IRM-8-13472 | SAIC | BWR | 0.00010 | ND | 0.01705 | ND | 0.01715 | 0.01259 | 0.01259 | 1 |
| LPRM-4-13031 | SAIC | BWR | .00000 | ND | 0.00008 | ND | 0.00008 | 0.00050 | 0.00050 | 2 |
| LPRM-4-13032 | SAIC | BWR | .00000 | ND | 0.00009 | ND | 0.00009 | 0.00032 | 0.00032 | 2 |
| LPRM-8-13470 | SAIC | BWR | 0.00004 | ND | 1.45000 | ND | 1.45004 | 0.00119 | 0.00119 | 1 |
| LPRM-14-1 | PNL | BWR | 0.34000 | 0.72273 | 1.11000 | 0.01900 | 2.19173 | 2.91429 | 2.91429 | 2 |
| LPRM-1-1 | SAIC | BWR | 0.00488 | 1.68182 | 1.40000 | 0.03167 | 3.11836 | 2.14286 | 2.14286 | 1 |
| LPRM-12-11582 | SAIC | BWR | 0.00168 | 0.07000 | 0.00950 | ND | 0.08118 | 0.10900 | 0.10900 | 2 |
| LPRM-14-2 | PNL | BWR | 0.39375 | 0.83636 | 1.29000 | 0.02203 | 2.54215 | 3.37143 | 3.37143 | 2 |
| LPRM-1-2 | SAIC | BWR | 0.00288 | 1.00000 | 0.08000 | 0.02800 | 1.11088 | 1.28571 | 1.28571 | 2 |
| LPRM-14-3 | PNL | BWR | 0.61500 | 1.30909 | 2.01000 | 0.03433 | 3.96842 | 5.27143 | 5.27143 | 2 |
| LPRM-6-1 | SAIC | BWR | 0.05250 | 1.72727 | 1.00000 | 0.00040 | 2.78017 | 2.57143 | 2.57143 | 1 |
| LPRM-5-1 | SAIC | BWR | 0.04000 | 2.95455 | 1.15000 | NA | 4.14455 | 4.85714 | 4.85714 | 2 |
| LPRM-6-2 | SAIC | BWR | 0.07250 | 2.36364 | 1.95000 | 0.00063 | 4.38677 | 3.42857 | 3.42857 | 1 |
| LPRM-5-2 | SAIC | BWR | 0.01875 | 1.40909 | 0.55000 | NA | 1.97784 | 2.28571 | 2.28571 | 2 |
| LPRM-4-13033 | SAIC | BWR | 0.00001 | ND | 0.00005 | ND | 0.00006 | 0.00055 | 0.00055 | 2 |
| POISON CUR.-11- | SAIC | BWR | 0.00026 | 0.12364 | 0.21500 | ND | 0.33890 | 0.19286 | 0.19286 | 1 |
| BPRA-13-A9P21 | WMG/RADCOR | PWR | 0.01913 | 0.05409 | 0.07400 | 0.00100 | 0.14821 | 0.19857 | 0.19857 | 2 |
| BPRA-13-12P27 | WMG/RADCOR | PWR | 0.01325 | 0.03727 | 0.05100 | 0.00068 | 0.10220 | 0.13700 | 0.13700 | 2 |
| BPRA-13-16P73 | WMG/RADCOR | PWR | 0.01073 | 0.03027 | 0.04125 | 0.00056 | 0.08280 | 0.11100 | 0.11100 | 2 |
| BPRA-13-16P78 | WMG/RADCOR | PWR | 0.00878 | 0.02477 | 0.03390 | 0.00045 | 0.06790 | 0.09200 | 0.09200 | 2 |
| BPRA-15-2 | WMG/RADCOR | PWR | 0.04925 | 0.13727 | 0.22950 | 0.00247 | 0.41849 | 0.50857 | 0.50857 | 2 |
| BPRA-7-2 | WMG/RADCOR | PWR | 0.01638 | 0.04332 | 0.07200 | 0.00056 | 0.13225 | 0.15286 | 0.15286 | 2 |
| BPRA-13-20P56 | WMG/RADCOR | PWR | 0.00728 | 0.02055 | 0.02800 | 0.00047 | 0.05629 | 0.07614 | 0.07614 | 2 |

## Table 4.3 Part 61 Classification Summary (continued)

| HARDWARE KIND | METHOD | PLANT TYPE | C-14/ LIMIT | Ni-59 / LIMIT | NB-94 / LIMIT | TC-99 / LIMIT | SUM OF THE TABLE 1 FRACS | NI-63 / LIMIT | SUM OF THE TABLE 2 FRACS | DETRMN. TABLE |
|---|---|---|---|---|---|---|---|---|---|---|
| BPRA-13-20P70 | WMG/RADCOR | PWR | 0.00891 | 0.02514 | 0.03430 | 0.00046 | 0.06881 | 0.09229 | 0.09229 | 2 |
| BPRA-13-B10P21 | WMG/RADCOR | PWR | 0.01550 | 0.04391 | 0.06000 | 0.00080 | 0.12021 | 0.16143 | 0.16143 | 2 |
| GUIDE TUBE15-1 | WMG/RADCOR | PWR | 0.09938 | 0.25727 | 0.42750 | 0.00300 | 0.78715 | 0.95143 | 0.95143 | 2 |
| GUIDE TUBE-7-0 | WMG/RADCOR | PWR | 1.51250 | 3.99545 | 5.80000 | 0.04700 | 11.35495 | 13.85714 | 13.85714 | 2 |
| PRIM SCR-13APS | WMG/RADCOR | PWR | 0.01275 | 0.03641 | 0.05050 | 0.00067 | 0.10033 | 0.13357 | 0.13357 | 2 |
| PSA-15-1 | WMG/RADCOR | PWR | 0.03375 | 0.09591 | 0.16550 | 0.00179 | 0.29695 | 0.35857 | 0.35857 | 2 |
| SEC SCR-13-SS4 | WMG/RADCOR | PWR | 0.01054 | 0.03009 | 0.04190 | 0.00055 | 0.08308 | 0.11171 | 0.11171 | 2 |
| RCCA-13-R123 | WMG/RADCOR | PWR | 0.00816 | 0.02314 | 0.03010 | 0.00037 | 0.06177 | 0.08457 | 0.08457 | 2 |
| RCCA-15-1 | WMG/RADCOR | PWR | 0.00043 | 0.00115 | 0.00253 | 0.00002 | 0.00414 | 0.00401 | 0.00401 | 1 |
| RCCA-7-1 | WMG/RADCOR | PWR | 0.00389 | 0.01745 | 0.03305 | 0.00077 | 0.05516 | 0.06543 | 0.06543 | 2 |
| THIMBLE PLUG-7 | WMG/RADCOR | PWR | 0.05375 | 0.22136 | 0.44950 | 0.00903 | 0.73365 | 0.79000 | 0.79000 | 2 |
| THIMBLE PLUG-7 | WMG/RADCOR | PWR | 0.14375 | 0.59091 | 1.20000 | 0.02413 | 1.95879 | 2.12857 | 2.12857 | 2 |
| THIMBLE PLUG13 | WMG/RADCOR | PWR | 0.00793 | 0.03664 | 0.05850 | 0.00150 | 0.10456 | 0.13557 | 0.13557 | 2 |
| THIMBLE PLUG-1 | WMG/RADCOR | PWR | 0.04188 | 0.20000 | 0.04015 | 0.00095 | 0.28298 | 0.73857 | 0.73857 | 2 |

## 5.0 CHARACTERIZATION METHOD UNCERTAINTIES

As discussed above there are two methods used to characterize irradiated hardware: the direct sample method and the activation analysis method. Both methods rely on a two-step approach which involves (1) determination of $^{60}$Co, and (2) determination of scaling factors for hard-to-detect Part 61 radionuclides and other radionuclides of interest. Both steps are equally important to ensure reasonably accurate characterization of radionuclides affecting classification.

The determination of $^{60}$Co concentration is important since it is used to scale the concentrations of the hard-to-measure radionuclides. Any error in $^{60}$Co concentration makes all other radionuclide concentrations suspect. The determination of scaling factors is also important because Part 61 classification is totally dependent on scaled radionuclide concentrations.

Both methods involve uncertainty in each major step. This section discusses those aspects of each characterization method which can result in uncertainty. The potential degree of uncertainty is also examined to the extent practicable from published data.

### 5.1 Direct Sample Method

For the direct sample method, hardware is characterized in two distinct phases. The first phase involves characterization of a representative component or batch of components using a combination of gamma spectroscopy, direct sampling and contact radiation profiling. The second phase involves application of the $^{60}$Co dose-to-curie factor and scaling factors obtained from the first phase to the bulk of the hardware to be characterized using contact radiation profiles. The elapsed time between these two phases can typically vary from about 6 weeks to about 6 months. The specifics of the technique are discussed in Appendix A.

The potential sources of uncertainty during the first phase include the following:

1. The efficiency of the gamma scan detector as applied to the component geometry.
2. The masking of potential variations in local activation due to the use of a simulated point source for the gamma scan measurement.
3. Radiation profiling conditions.

During the second phase, the results of the initial characterization phase are applied to the entire hardware inventory and all components to be characterized are profiled. The potential sources of uncertainty during this phase include:

1. Representativeness of the radiochemical analysis results.
2. Differences in the radiation profiling conditions between the first and second phases.

The potential sources of uncertainty during both phases of a typical project using the direct sample method are discussed below.

### 5.1.1 Direct Sample Method - $^{60}$Co Determination

This method combines a measured gamma scan reading with a contact dose level reading to determine a component specific $^{60}$Co dose-to-curie conversion factor in units of Ci of $^{60}$Co per $m^3$ per R per Hr. Measurements are taken at a point location on a pre-selected component using material composition information to represent all similar components in a batch or lot. The calculated dose-to-curie conversion factor is then used in conjunction with the component volume, a constant for similar types of components, and contact radiation profiles to determine the axial distribution of $^{60}$Co in the pre-selected component. The potential sources of uncertainty associated with these measurements are presented below.

#### 5.1.1.1 Gamma Scan Detector Efficiency

The SAIC direct sample method involves placement of a collimator tube connected to a gamma scan detector at a single pre-selected location on a hardware component to measure the gamma emitting radionuclides in Ci per $m^3$. Except for fuel channels where $^{125}$Sb is present and cluster control assemblies where $^{110m}$Ag is present, $^{60}$Co is the principal radionuclide measured. The contact radiation level reading taken later at the same location with an RO-7 ionization chamber is used in conjunction with the gamma scan reading to obtain a $^{60}$Co dose-to-curie conversion factor in units Ci of $^{60}$Co per $m^3$ per R/Hr. The potential uncertainty in this measurement is related to the efficiency and calibration of the gamma scan detector.

Gamma-ray spectrometers determine the relative concentration of gamma-emitting radionuclides in a source by counting the number of gammas seen at a distribution of energies. To determine the absolute concentration of gamma-emitting radionuclides, the efficiency of the detector must be determined for the geometry of the source, the gamma-ray energies emitted by the source, and the relative location of the detector to the source.

SAIC has calculated detection efficiencies for source geometries that include liquid, gas, disk, pipes and solid sources in large (>30,000 gallon) and small (5 cc) right-circular cylinders, spheres, compacted solids (ore and charcoal) and rectangular configurations. The calculations account for gamma-ray absorption in the source material, the sample container, and any external absorbing material. The referenced SAIC literature does not indicate that calibrations have been performed underwater, mimicking the spent fuel pool environment and source geometry found in irradiated hardware.

Using National Bureau of Standards (NBS) mixed point source standards, SAIC has experimentally determined the on- and off-axis efficiencies for the SAIC system at various source-detector distances. From these experiments, mathematical models of the detector efficiency for various source geometries and source-detector configurations have been developed. The referenced SAIC reports initially indicated the gamma scan system could determine $^{60}$Co concentrations to an accuracy of within +/- 40 %(1). The most recent SAIC report(13) indicates this error has been reduced to ± 20%.

The most recent SAIC report also indicates that although efforts are made to ensure contact, there is a potential error associated with the placement of the collimator tube on the component. Distances between the collimator and the component of up to 1/2 inch can result in an additional 10 % error in the gamma scan reading(13).

### 5.1.1.2 Masking of Local Variances

The collimator is placed on the component at a single location. The collimator is approximately 1 inch in diameter, and is usually located on the flat surface of components like control rod blades and flow channels, and on the round surface of tube-like components. This can lead to uncertainties related to localized variances in the $^{60}$Co concentrations in components which contain strong neutron absorbers. These include BWR CRB's and poison curtains, and PWR reactor cluster control assemblies and burnable poison rod assemblies.

One example of this effect is a BWR control rod blade. The collimator is placed at a point on the flat of the blade, looking at a 1-inch diameter spot out of a total blade width of approximately 5 inches. This area of the blade consists of thin tubes containing boron carbide pellets covered with a thin stainless sheath. At the center of the blade is the relatively heavy stainless steel hub. The collimator measurement of $^{60}$Co depends on its radial location relative to the radial variance in the stainless steel mass and the neutron flux variance from the CRB hub to the blade tip.

The stainless steel mass varies significantly from solid stainless (7.8 g/cc) at the hub, to an equivalent metal density of about 4 g/cc in the flat containing the boron carbide pellets. Thus, the stainless mass at any point along the blade flat is 50 percent of that at the hub. The CRB cross section shows that the hub extends about 0.8 inch from the axial center and represents about 30 percent of the metal cross sectional area. The blade flat containing boron carbide pellets extends from the hub to the blade tip, a distance of about 4 inches, and represents about 70 percent of the metal cross sectional area. Discrete models of a CRB cross section show that the hub section contributes about 30 percent of the dose rate seen six inches from the CRB axis. Placement of the collimator tube on the blade flat neglects the effect of the hub, and thus underestimates the $^{60}$Co content. The extent of this potential underestimation cannot, however, be quantified.

The presence of the boron carbide pellets in the blade flat results in a flux depression across the blade. This flux depression can decrease the activation of the stainless in the blade flat by as much as 50 percent relative to the activation of the stainless at the hub.

The CRB is surrounded by four fuel assemblies, and the power ratios of the fuel pins adjacent to the CRB also vary from the CRB hub to the blade tip. Fuel pins adjacent to the hub can have a power ratio as low as 0.4, while those at the tip can have a power ratio as high as 1.2. On the average, the difference in power ratios between the CRB hub and CRB blade tip is about 100 percent. Thus, activation of the CRB near the hub can be 50 percent lower than activation at the blade tip. The effect of the adjacent fuel pin power ratios partially offsets the effect of flux depressions due to the presence of boron carbide pellets.

The combined effect of these flux variance uncertainties cannot be quantitatively determined. However, measurements of boron depletion as a function of distance from the CRB hub to the blade tip have been performed(14). These measurements indicate that boron depletion at the midpoint of the blade is about 25 percent less than depletion at the blade ends. This would indicate that placement of the collimator in the mid-section of the blade could underestimate $^{60}$Co activity by about 10 to 15 percent.

Detailed radial measurements of CRB's could help quantify the effect of both these potential uncertainties.

BWR control rod blades represent the worst case conditions for local variations; the extent of this potential uncertainty should be less for other components containing neutron absorbers.

### 5.1.1.3 Radiation Profiling Instruments

The direct sample method relies on an RO-7 ionization chamber to obtain contact axial radiation profiles. These measurements are taken during both phases of a typical project and the potential sources of uncertainty differ for each phase.

During the initial phase of a project, a contact dose rate measurement is obtained at the location where the gamma scan collimator tube is placed. The reading obtained at this location is used to convert the local gamma scan reading of Ci per $m^3$ to Ci per $m^3$ per R/Hr. The potential sources of error include instrument calibration and source-detector distance variances. The most recent SAIC report(13) indicates that the manufacturer routinely calibrates the instrument in air with an NBS $^{137}$Cs source and that the accuracy is better than 5% over the range of 0.16 to 320 R/Hr. This report also indicates that the manufacturer believes that the accuracy at the high end of the range, 20,000 R/Hr maximum, is also better than 5%. Other manufacturer literature(15) indicates that the 5% uncertainty applies to both the low (to 2 R/hr) and mid ranges (to 200 R/hr), but a 10% uncertainty applies to the high range probe (to 20,000 R/hr). Since contact radiation readings on most components, except fuel flow channels, are in the high range, the 10% uncertainty would generally apply. This material also indicates that when the instrument is calibrated in air, and then placed in a housing for underwater use, there is a tendency to read low by about 5%.

Figure 5.1 shows the dose rate vs. distance for a PWR thimble plug, BWR local power range monitor, and BWR poison curtain. During the initial phase, considerable care is taken during component profiling to ensure that contact between the component and the detector is maintained. However, as shown from Figure 5.1, a very slight variance in source-detector distance, on the order of 1/2 inch, can affect the radiation reading by 20 to 30 percent depending on the component type. During the initial phase this uncertainty is not absolute since the reading taken is used relative to the gamma scan reading. Thus, the extent of the uncertainty depends on the RO-7 location relative to the collimator tube location. If the collimator tube is 1/2 inch off the component and the RO-7 ionization chamber is also 1/2 inch away, the $^{60}$Co dose-to-curie conversion factor could be overestimated by 20%. If the collimator tube is in contact and the

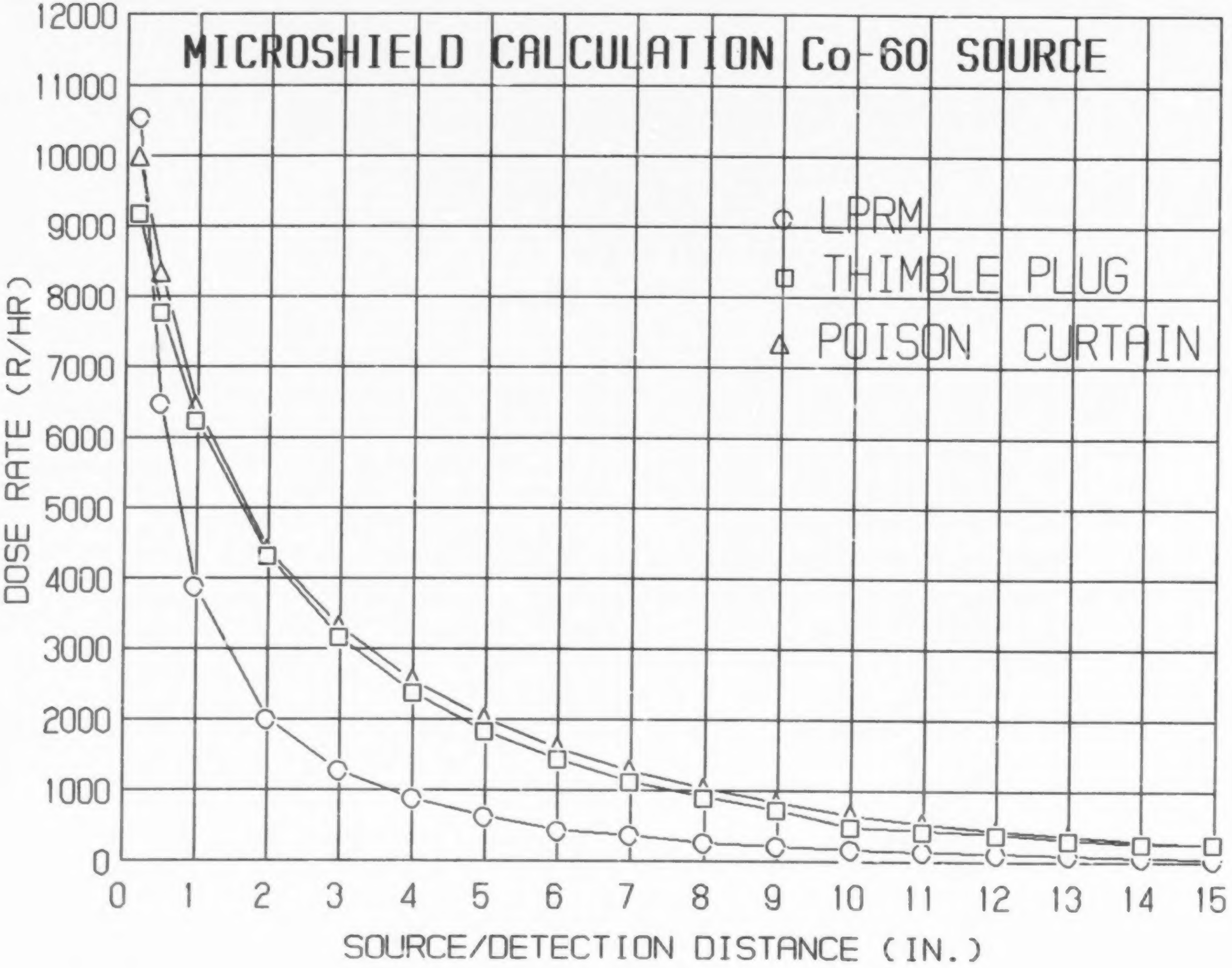

Figure 5.1 Effect of Dose Rate and Distance From Source

RO-7 is 1/2 inch off, the $^{60}$Co dose-to-curie conversion factor could be overestimated by about 30 percent. Conversely, if the collimator tube is off by 1/2 inch and the RO-7 is in contact, the $^{60}$Co dose-to-curie conversion factor could be underestimated by about 10 percent.

Most radiation profiling is performed during the final phase of a project. These profiles are generally performed by a different organization and are not necessarily taken with the same detector, using the same procedures. The $^{60}$Co dose-to-curie conversion factor has already been fixed by the initial measurement, so potential uncertainties result from ensuring that the component and detector are in contact. Readings other than those at full contact seem likely due to the production nature of this phase. As shown on Figure 5.1, the uncertainties associated with distance variances between 1/2 inch and 1 inch are between 20 and 50 percent dependent on the component type. The lower radiation levels measured under these conditions will result in underestimation of the $^{60}$Co content.

### 5.1.2 Direct Sample Method-Scaling Factor Determination

When the local gamma scan reading is taken, a bulk metal sample, weighing between 10 and 100 milligrams, is taken at essentially the same location as the scan. This bulk metal sample is radiochemically analyzed to obtain scaling factors for Part 61 radionuclides by determining the relative content of the Part 61 radionuclides and $^{60}$Co in the sample. These analyses are usually not performed for $^{59}$Ni. A generic $^{63}$Ni based scaling factor is used for this Part 61 radionuclide.

#### 5.1.2.1 Sample Representativeness

The method uses a relatively small representative sample to characterize a single pre-selected component and then applies the results obtained for the representative component to all components of the same lot or batch. One of the assumptions implicit in this method is that the relative compositions of constituents are uniform throughout the component. This is reasonable except for the potential hot spot variances on certain types of components as described above. The other assumption is that the sampled component is representative for all the components that will be characterized using the same dose-to-curie conversion factor and scaling factor set.

A review of the referenced SAIC reports does not indicate the basis for selection of this single component. However, these reports do indicate that components of different types are grouped together into individual batches. Depending on the different irradiation histories and cooling times among the components which comprise a batch, the potential uncertainties could be considerable. As discussed in Section 3.1, scaling factors can be significantly affected by these parameters. As shown on Figure 3.2, a 2 year difference in irradiation time can affect the $^{63}$Ni scaling factor by about 10 percent. The effect of cooling time differences is much greater, as shown by Figure 3.4. The same two year difference in cooling time can affect the $^{63}$Ni scaling factor by about 25 percent. The selection of components according to irradiation history and cooling time to define a batch for direct sampling would essentially eliminate this source of uncertainty.

### 5.1.2.2 Radiochemical Analysis of Direct Sample Content

The accuracy of the radiochemical analysis of the bulk metal sample is dependent on the laboratory procedures employed and the radionuclide of interest. Those radionuclides which exist in irradiated hardware in relatively large concentrations have relatively small uncertainties. These radionuclides include $^{14}$C, $^{60}$Co and $^{63}$Ni. For $^{94}$Nb and $^{99}$Tc, the matter is quite different. These radionuclides exist in activated metals near or below the minimum detectable limits (MDL) of the chemical analyses. Additionally, analyses are no longer performed for $^{59}$Ni. Generic scaling factors for $^{59}$Ni are obtained from NUREG/CR-4101(16).

Since the samples are analyzed to obtain Part 61 radionuclide content relative to $^{60}$Co, the degree of uncertainty associated with radiochemical analysis of $^{60}$Co affects the scaling factor determination. This uncertainty is on the order of 5 to 10%(13). For radionuclides such as $^{63}$Ni and $^{14}$C, this uncertainty is reported to be on the order of 10%(13). These combined potential variances in the radiochemical analyses results can lead to $^{63}$Ni and $^{14}$C scaling factor variances of up to 15 percent. If $^{60}$Co is high, and $^{63}$Ni is low, the scaling factor can be underestimated by this amount. The opposite result is obtained when $^{60}$Co is low and $^{63}$Ni is high.

Where LLD values are obtained from analysis, usually the case for $^{94}$Nb and $^{99}$Tc, the uncertainty can be much larger. Since use of LLD values overestimates the radionuclide content in the component, their use is a conservative approach for the purposes of Part 61 classification.

From the data base presented in Section 4, it appears that uncertainties for $^{59}$Ni can also result when generic scaling factors are used. These data indicate that the reported $^{59}$Ni scaling factors are higher than those calculated by the activation analysis methods.

## 5.2 Activation Analysis Method

The activation analysis method also involves two phases. The first phase develops a component-specific data base which describes the physical characteristics of each type of component to be characterized, plus the irradiation history of each batch of components to be characterized. The second phase uses this component-specific data plus non-contact radiation profiles to characterize each component. The elapsed time between these phases is typically several weeks. The specifics of three different techniques are discussed in Appendix A. The uncertainties associated with each phase of a typical project are discussed below.

The first phase defines the project and component specific data that will be used to perform activation analysis calculations. The sources of uncertainty during this phase include:

1. Accuracy of component dose-to-curie models.
2. Accuracy of irradiation history and cooling time data.
3. Accuracy of the neutronics data base.
4. Materials composition in base metals.

The second phase applies the calculated dose-to-curie conversion factors to component-specific data, to characterize each component using non-contact radiation profiles. The potential sources of uncertainty during this phase include:

1. Accuracy of radiation profiling instruments.
2. Variances in radiation profiling conditions.

These potential uncertainties are discussed below. As discussed in Section 3.0, a number of different techniques are used. No attempt has been made to distinguish between these techniques, since the uncertainties are generally method dependent.

### 5.2.1 Determination of $^{60}$Co

Most activation analysis techniques rely on a computer code calculated model of a component to obtain a $^{60}$Co dose-to-curie conversion factor in units of either Ci per $m^3$ per R/Hr or Ci per R/Hr. This factor is then used with non-contact radiation measurements to determine the $^{60}$Co content of the component at the time of the radiation measurement. Potential sources of uncertainties associated with $^{60}$Co determination are discussed below.

#### 5.2.1.1 Accuracy of Dose-to-Curie Models

Activation analysis methods employ calculated dose rate-to-curie conversion factors to estimate the $^{60}$Co content of hardware. These conversion factors are generated using point-kernel shielding codes for the various hardware geometries. Different computer codes and models are used. TPT employs the PATH(5) computer code to perform these calculations. WMG uses the MICROSHIELD(3) computer code, derived from the ISOSHLD(17) computer code and ported to run on personal computers. PATH accommodates a single radioactive source as does MICROSHIELD. PNL utilizes the SLAB(10) point-kernel shielding code which allows for multiple sources. There are two sources of uncertainty associated with the use of these codes: (1) code-calculated accuracy, and (2) source content.

The computed results accuracy depends on the assumptions made to perform the calculations and the simplified model applied to the component geometry. For simple geometries, the codes can generally predict $^{60}$Co concentrations within 25%. The model uncertainties are, however, component specific. Normally, a geometrically simplified model of the radioactive source is established as input for the code. A unit curie content of $^{60}$Co (one curie total for the source or one curie per cubic centimeter, for example) is used as the radioactive source. Exposure-rate calculations are performed with water shielding at source-detector distances typically between 6 and 12 inches. For simple geometries like LPRM's, poison curtains, and channels, the 25% figure is reasonable. For more complex geometries like CRB's, thimble plug assemblies, and cluster assemblies, the accuracy is also dependent on the models used as code input.

To examine the effect of source geometry simplification, a complex geometry, that of a PWR thimble plug, was analyzed. The thimble plug was modeled as a discrete and as a homogenized source. Calculations were performed under three sets of conditions at a source-detector distance of 6 inches in water. The calculated $^{60}$Co dose-to-curie conversion factors were as follows:

(1) Discrete MICROSHIELD calculation - 0.299 Ci per R/hr,

(2) Homogenized MICROSHIELD calculation - 0.348 Ci per R/hr,

(3) Discrete QADCGGP(18) calculation - 0.327 Ci per R/hr.

For this component, these results indicate that both discrete model calculations give reasonably consistent results - within 10%. Homogenization of this source also results in an overestimation of the factor by about 25% using MICROSHIELD. The effect of geometry was also found to depend on distance from the source. The further away from the source, the better the discrete and homogenized models agreed. Thus, it appears that the use of homogenized source geometries can lead to an overestimation of $^{60}$Co content by about 25% at source detector distances of 6 inches. Refinement in component geometric models could reduce this source of uncertainty.

Another source of uncertainty in the dose-to-curie conversion factor lies in the assumption that all of the gamma radiation seen at the detector will be $^{60}$Co. For most hardware, the gamma exposure at the detector 6 to 12 inches from the source underwater, will be primarily due to $^{60}$Co; the effect of other radionuclides will be less than 5%. There are, however, several other gamma emitters that should be considered on a project and component-specific basis.

Where manganese is present in the base metal and the hardware has recently been removed from the core, $^{54}$Mn can generally contribute 5-10% to the dose rate and should be considered. Neglecting the $^{54}$Mn contribution overestimates the $^{60}$Co content.

For Zircaloy components, activation of the trace element tin in the base metal will create $^{125}$Sn and $^{125m}$Sn, which decay rapidly to $^{125}$Sb. The $^{125}$Sb remaining has a complex gamma decay scheme of gamma energies less than 0.62 Mev. The effect of $^{125}$Sb in the source was examined by performing a series of calculations. These calculations show that for a source containing one curie of $^{60}$Co and one curie of $^{125}$Sb, the dose rate through 6 inches of water from the $^{125}$Sb is about 25 percent of the total dose rate seen by the detector. Neglecting the gamma dose rate due to $^{125}$Sb will overestimate the $^{60}$Co concentration. The dose-to-curie conversion factor calculation should therefore include the $^{125}$Sb gamma contribution.

For components containing the neutron-absorbing material silver-indium-cadmium, irradiation of the silver creates the activation product $^{110m}$Ag. The effect of $^{110m}$Ag was also examined. Calculations show that for a source containing one curie of $^{60}$Co and one curie of $^{110m}$Ag, the $^{110m}$Ag dose rate through 6 inches of water is about 60 percent of the total dose rate seen by the detector. Neglecting the $^{110m}$Ag gamma dose will overestimate the $^{60}$Co concentration. The dose-to-curie conversion factor calculation should therefore include the $^{110m}$Ag gamma contribution.

As the source detector distance is increased beyond 6 inches, the relative contributions of these three radionuclides to the measured dose rate decrease.

### 5.2.1.2 Radiation Profiling Instruments

Different instruments have been used to take radiation measurements. These instruments include RO-7 ionization chambers, high and low range CP-MU's, and Dositec DR-10's, PR-7's and PR-2's. Since the determined $^{60}$Co content is directly proportional to the instrument reading, instrument accuracy has a direct effect on the potential uncertainty.

#### Eberline RO-7

This instrument is widely used and its accuracy is discussed in Section 5.1.1.3 above. The major difference in its application for activation analysis is that the non-contact radiation level readings are generally lower by factors of 3 to 10. Thus, the radiation levels are well below the 20,000 R/Hr maximum for the instrument. The accuracy in the range of interest is still +/-10%.

#### CP-MU

These instruments are ion detection chambers designed for underwater use. Both the high range (DMU-1000) and the mid range (DMU-1) probes have been used for hardware projects. There are four linear ranges on the CP-MU readout, so that with both probes, levels between 100 mrem/hr and 1 million R/hr can be read. A separate set of high range calibrators which cover four ranges ($10^3$ R/hr to $10^6$ R/hr) are available. The manufacturer's literature(19) indicates accuracy of +/-15% of full scale.

#### Dositec

These instruments have been used in several projects and have also been extensively evaluated in underwater applications. There are several models. The PR-2 series reads any three ranges from 5 R/hr to 50,000 R/hr, and the PR-7 reads 7 ranges to 50,000 R/hr. The DR-10 series reads any three ranges and includes an integrated dose alarm. Dositec has conducted extensive field evaluations of instrument performance(20). Underwater dose rate measurements were compared to data obtained from TLD measurements, pocket ion chambers, and a condenser R-meter from another vendor. Additionally, Dositec compared readings between survey meters of the same model and between different models they manufacture. The Dositec study reported differences of up to 5% between the Model DR-10 and TLD data with the DR-10 measurement tending to be less than the TLD measurement. Generally, the manufacturer believes all the instruments are accurate to with +/-10% of full scale(21).

#### 5.2.1.3 Variances in Profiling Conditions

Long components are profiled along their axial lengths to determine the axial distribution of $^{60}$Co. Shorter components, such as thimble plug assemblies, are usually measured at a single location to obtain $^{60}$Co content for the entire component. Uncertainty arises from potential differences in the component-detector distance relative to the detector distance used for the dose-to-curie conversion model.

Figure 5.1 shows dose rate vs. distance calculations for a PWR thimble plug, a BWR local power range monitor, and a BWR poison curtain. Generally, care is taken during component profiling to ensure that a consistent component-detector distance is maintained. However, as shown from these figures, at source detector distances of 6 to 12 inches, variances in source-detector distance, on the order of one inch, can affect the radiation reading by 20 to 25 percent depending on the component type.

Where the reading is taken closer than the distance upon which the dose-to-curie conversion factor is based, the $^{60}$Co content is overestimated. Conversely, readings taken at distances greater than the base distance underestimate the $^{60}$Co content.

### 5.2.2 Determination of Scaling Factors

Parameters which affect Part 61 radionuclide scaling factors include relative materials composition, relative cross sections and neutron fluxes, component irradiation history, and cooling time. Most activation analysis methods decay correct the $^{60}$Co content at the time of radiation measurement to the time of core discharge, determine scaling factors using the relative activation of the Part 61 radionuclide and $^{60}$Co at discharge, and then decay correct all concentrations to the time of shipment. Potential uncertainties in the scaling factors determined using this method are discussed below.

#### 5.2.2.1 Material Compositions In Base Metal

Proper performance of an activation analysis requires accurate materials composition data to obtain the relative relationships between $^{60}$Co concentrations and Part 61 radionuclide concentrations.

Material composition data for steels may be obtained from a number of sources. The source most commonly referenced is the ASTM Standards(22). Unfortunately, the ASTM Standards do not contain data for many of the trace elements which are precursors to Part 61 radionuclides These trace elements include nitrogen, cobalt, niobium and technetium. Consequently, data is often obtained from other sources, i.e., NUREG/CR-3474, Stephens and Pohl(23), and the "Handbook of Stainless Steels"(24).

A review of these sources illustrates the wide range of concentrations predicted for precursor elements. Compositions may vary by a factor of two for those precursor elements for which chemical compositions are not well known. Where heat data is not available, material compositions for an activation analysis are usually chosen from the ASTM standards. The potential variances in base metal composition for Part 61 precursor elements is discussed below.

> Cobalt - In most metals used in reactors (stainless steel and inconel) cobalt exists as an impurity. Very often the cobalt concentration is unknown. Metal heat chemistry data may be available, and the values most often seen in material certifications for cobalt concentrations range between 0.05 and 0.20 weight percent (w/o).
>
> In NUREG/CR-3474, chemical analysis for eight Type 304L stainless steel samples are reported. Results show an average cobalt concentration of approximately 0.14 w/o. Additionally, two samples of inconel 718 were analyzed, and found to contain somewhat lower cobalt concentrations. In the absence of heat data, most activation analysis techniques use a cobalt content of 0.10 w/o for Type 304 stainless steel and inconel components. Although lower than the NUREG/CR-3474 average, the 0.10 w/o cobalt concentration leads to more conservative Part 61 scaling factors.
>
> Nickel - Elemental nickel is of particular importance to Part 61 classification. According to ASTM specifications for Type 304 stainless steel, nickel concentration may vary between 8 w/o and 11 w/o. Typically a conservative value of 10 w/o is used. For the various inconel alloys, nickel concentrations may vary between 52 w/o and 78 w/o. Typically, the nominal ASTM value for a particular inconel alloy is used.
>
> The uncertainties in the scaling factors for $^{63}$Ni and $^{59}$Ni are a direct function of the elemental composition ranges for nickel and cobalt. For Type 304 stainless steel, the 10 w/o value for nickel composition may overestimate the nickel scaling factors by 25%, or underestimate them by about 10%. Increasing the cobalt concentration beyond the 0.10 w/o value will make the nickel scaling factor less conservative. Decreasing the cobalt concentration will make the nickel scaling factor more conservative.

For inconels, the nominal ASTM values which allow for a few percent variation in nickel concentrations are normally used. Due to the relatively high nickel content, the range of uncertainty in the nickel concentration is relatively small, a few percent. However, the scaling factors will vary with the cobalt concentrations.

Niobium - Irradiation of niobium impurities in the base metal produces $^{94}$Nb. Since a niobium concentration is not specified in the chemical requirements for Type 304L stainless steel, significant variances may occur in base metal concentrations. Typically, for activation analyses, a niobium concentration of 50 ppm is chosen as the base metal concentration. NUREG/CR-3474 reports niobium concentrations ranging between 5 and 300 ppm, with an average value of approximately 90 ppm.

The difference between the 50 ppm typically used and the NUREG/CR-3474 average of 90 ppm indicates that the use of 50 ppm can underestimate the scaling factor by up to 80% in stainless steel components. Using the NUREG maximum value of 300 ppm increases this uncertainty to a factor of 6. These apparent underestimates are increased with cobalt concentrations below 0.10 w/o. Thus, the NUREG data would indicate that the use of 50 ppm consistently underestimates the $^{94}$Nb scaling factor. However, the available data shown in Figures 4.3 and 4.4, and discussed in Section 4.0, indicates that both the direct sample method and the activation analysis methods use conservative $^{94}$Nb scaling factors. Better materials composition data on niobium would reduce this uncertainty due to materials composition and would also resolve the apparent discrepancy concerning use of 50 ppm niobium.

For inconels, the average alloy value is used. For example, the ASTM chemical requirements for inconel-718 specify niobium concentrations between 4.75 and 5.50. At 10 w/o cobalt, the total scaling factor uncertainty for inconel-718 is about 15% or +/- 7.5 %. This uncertainty will increase or decrease dependent on the cobalt concentration as discussed above.

Molybdenum - The presence of $^{99}$Tc in irradiated metal is due to activation and subsequent beta decay of $^{98}$Mo in the base metal. In Type 304 stainless steel, molybdenum exists as an impurity; there is no chemical composition requirement for it in the ASTM Specifications. NUREG/CR-3474 reports molybdenum concentrations in Type 304 stainless steel between 80 and 5500 ppm, with an average value of 2600 ppm. Activation analyses generally use 500 ppm for molybdenum compositions.

The difference between the 500 ppm typically used and the NUREG/CR-3474 average of 2600 ppm indicates that the use of 500 ppm can underestimate the scaling factor by a factor of about 5. These apparent underestimates are increased with cobalt concentrations below 0.10 w/o. Thus, the NUREG data would indicate that the use of 500 ppm consistently underestimates the $^{99}$Tc scaling factor. However, the available data shown in Table 4.2 indicates that the $^{99}$Tc scaling factors obtained from both the direct sample method and the activation analysis methods are comparable. Better materials composition data on molybdenum would reduce this uncertainty due to materials composition.

For inconels, average alloy compositions are typically used. For example, the average ASTM chemical requirements for inconel-718 have a molybdenum composition range of 2.80 to 3.30 w/o. At 0.10 w/o cobalt, the average value leads to uncertainties of about +/- 8% in the $^{99}$Tc scaling factor. This uncertainty will increase or decrease dependent on the cobalt concentration as discussed above.

Nitrogen - Nitrogen is a trace impurity in most hardware materials. It is activated by neutrons to form $^{14}$C. NUREG/CR-3474 reports nitrogen concentrations between 400 to 525 ppm in Type 304 stainless steel. A value in the 400 ppm range is typically used for activation analyses. Thus, the $^{14}$C scaling factor could be underestimated by up to 10%. This uncertainty will increase or decrease dependent on the cobalt concentration as discussed above. Nitrogen data for inconels are not reported in NUREG/CR-3474.

### 5.2.2.2 Neutronics Data

Potential scaling factor uncertainties also depend on relative neutron activation cross sections and relative neutron fluxes in the various energy groups used for an activation calculation. The thermal and fast cross sections for the precursor radionuclides of interest are reasonably well-known and accepted, minimizing neutron cross sections uncertainties.

The scaling factor for a particular radionuclide is essentially equal to activation of the precursor radionuclide and the decay of the radionuclide, divided by activation and decay of cobalt. When a two-energy group calculation is performed, neutron fluxes for a fast and a thermal energy group are required as input. Since all Part 61 radionuclides of interest have relatively long half-lives compared to $^{60}$Co, this relationship can be illustrated by the following expression:

$$\frac{\text{Cross section(N)t*Flux(N)t + Cross Section(N)e*Flux(N)e}}{\text{Cross section(C)t*Flux(C)t + Cross Section(C)e*Flux(C)f}}$$

Where "N" designates the radionuclide of interest ,"C" the $^{60}$Co, "t" the thermal flux, and "e" the epithermal flux. Since the flux terms appear in both the top and bottom terms of this scaling factor ratio, the magnitude of the fluxes tends to cancel. However, since the group-dependent cross sections for the radionuclide of interest and $^{60}$Co differ, the ratio of the fast to thermal flux becomes important.

The effect of flux variances on scaling factors was examined. This examination was based on a thermal flux of 1E+13 n/cm$^2$/sec. The epithermal flux was increased a factor of two, from 2.5E+13 to 5.0E+13 n/cm$^2$/sec, to determine sensitivity to major variations in flux ratio's. The effects of varying the flux ratio on scaling factors were as follows:

| Radionuclide | Effect of Flux Ratio Variance |
|---|---|
| $^{63}$Ni | 15% scaling factor decrease |
| $^{59}$Ni | 15% scaling factor decrease |
| $^{94}$Nb | 30% scaling factor increase |
| $^{14}$C | 10% scaling factor decrease |
| $^{99}$Tc | 35% scaling factor increase |

Even though a factor of two difference in flux ratio's is considerable, the effect on scaling factors is relatively small. As shown for the case examined, the scaling factors for $^{94}$Nb and $^{99}$Tc are increased by 30-35%, and those for the nickels and $^{14}$C are decreased by 10-15%. Generally, neutron fluxes are taken from core physics data and the uncertainty in flux ratios should be lower than the factor of two considered. This will reduce these potential uncertainties. This effect could however be more pronounced for neutron-absorbing components like CRB's. For these components, the thermal flux is relatively depressed in the region near the neutron absorbing material, and the fast flux is depressed in the region near the thick pieces of metal. However, since whole components are characterized, these competing effects will tend to cancel each other.

### 5.2.2.3 Irradiation History and Cooling Time Data

As discussed in Section 3.1, proper performance of an activation analysis requires accurate irradiation history and cooling time data to obtain the relative relationships between $^{60}$Co concentrations and Part 61 radionuclide concentrations as a function of time. Generally, this information is maintained by the plant core engineering group, is available for most components, and is reasonably accurate. Where component specific histories are not available, a conservative combination of irradiation time and cooling time is applied to the component or lot of components. Thus, where histories are available, the uncertainty is a few percent at most. Larger uncertainties occur when histories are not available; however, in these cases, conservative assumptions are used for the analysis.

### 5.3 Potential Uncertainty Summary

The method specific uncertainties discussed above are summarized below. Uncertainties are divided into two categories; (1) random uncertainties which can result in underestimation or overestimation of the listed parameter, and (2) systematic uncertainties which either consistently underestimate or overestimate the parameter. The effect of the potential uncertainties on Part 61 classification is also discussed.

### 5.3.1 Direct Sample Method Uncertainty Summary

Potential uncertainties for the direct sample method are summarized in Table 5.1. The cumulative random uncertainties associated with $^{60}$Co determination are between 55% and 70% dependent on the local variance effect. The systematic uncertainties are phase dependent and are about -35% during phase one and from -35% to -55% during phase two.

The uncertainties in scaling factors are either random or conservative. The largest potential source of uncertainty, estimated to be +/-40%, is the representativeness of the sampled component relative to other components that will be characterized with the sample results. The uncertainties in the radiochemical analysis results are relatively small, ±15%. The use of LLD's for $^{94}$Nb, $^{99}$Tc and the generic $^{59}$Ni/$^{63}$Ni relationship overestimate scaling factors and are thus conservative.

The potential to consistently underestimate $^{60}$Co content by 35 to 55% would lead to underestimates of the concentrations of scaled radionuclides, and could affect classification. This is partially offset by the conservative scaling factors used for the Table 1 radionuclides $^{94}$Nb and $^{59}$Ni. Method uncertainty can be improved in two major areas: (1) ensuring use of consistent practices and equipment during phases one and two to minimize the systematic $^{60}$Co uncertainty and (2) consideration of component irradiation histories and cooling times in the phase one sampled component selection process to reduce the random uncertainties.

### 5.3.2 Activation Analysis Methods Uncertainty Summary

The potential uncertainties for the activation analysis methods are summarized in Table 5.2. The cumulative random

Table 5.1 Direct Sample Method Potential Uncertainties

| Source | Potential Range | Effect on Parameter |
|---|---|---|
| $^{60}$Co Determination | | |
| 1. Gamma scan calibration | +/-20% | Random |
| 2. Local variances (Neutron absorbers only) | +/-15% | Random |
| 3. Collimator position | -10% | $^{60}$Co Underestimate |
| 4. RO-7 calibration | -5% | $^{60}$Co Underestimate |
| 5. RO-7 Accuracy | +/-10% | Random |
| 6. Relative position of RO-7 and collimator | +/-25% | Random |
| 7. Local phase 1 profiling RO-7 vs component | -20% | $^{60}$Co Underestimate |
| 8. Phase 1 versus phase 2 profiling | -20 to -50% | $^{60}$Co Underestimate |
| Scaling Factor Determination | | |
| 9. Sample Representativeness | varies | Random |
| (a) irradiation history | +/-10% | |
| (b) cooling time | +/-30% | |
| 10. Radiochemistry | varies | Random |
| (a) $^{63}$Ni | +/-15% | Random |
| (b) $^{14}$C | +/-15% | Overestimate |
| 11. Use of LLD Value for $^{99}$Tc | NA | Overestimate |
| 12. Use of LLD Value for $^{94}$Nb | NA | Overestimate |
| 13. Use of Generic $^{59}$Ni Factor | NA | Overestimate |

## Table 5.2 Activation Analysis Method Potential Uncertainties

| Source | Potential Range | Effect on Parameter |
|---|---|---|
| 60Co Determination | | |
| 1. Dose to curie model | | |
| (a) Code accuracy | +/-25% | Random |
| (b) Source Geometry | to +25% | $^{60}$Co Overestimate |
| (c) Source content(1) | +5% to 60% | $^{60}$Co overestimate |
| 2. Instrument Accuracy | Varies | |
| (a) RO-7 | +/-10% | Random |
| (b) CP-MU | +/-15% | Random |
| (c) Dositec | +/-10% | Random |
| 3. RO-7 calibration | -5% | $^{60}$Co Underestimate |
| 4. Relative position of instrument/component | +/-25% | Random |
| 5. Component Cooling Time | +/-5% | Random |
| Scaling Factor Determination | | |
| 6. Materials Composition | Varies | Varies |
| (a) $^{63}$Ni | -10% +25% | Random |
| (b) $^{59}$Ni | -10% +25% | Random |
| (c) $^{94}$Nb | -80% minimum | Discrepancy-data show overestimate |
| (d) $^{14}$C | -10 % | Underestimate |
| (e) $^{99}$Tc | under by factor of 5 | Discrepancy-data show comparable to direct sample |
| 7. Neutronics Data(2) | Varies | Varies |
| (a) $^{63}$Ni | -15 % | Underestimate |
| (b) $^{59}$Ni | -15 % | Underestimate |
| (c) $^{94}$Nb | +30 % | Overestimate |
| (d) $^{14}$C | -10 % | Underestimate |
| (e) $^{99}$Tc | +35 % | Overestimate |
| 8. Irradiation History | Varies | Overestimate |

Notes:

(1) Nuclide and component specific (see section 5.2.1.1)

(2) Based on flux ratio variance of two (see section 5.2.2.2)

uncertainties associated with $^{60}$Co determination are between ±70% to ±75% dependent on the instrument used. Model homogenization results in a systematic overestimate of up to 25% which is partially offset by a 5% underestimate when an RO-7 instrument is used. There are also systematic uncertainties which overestimate $^{60}$Co content due to component specific radionuclides other than $^{60}$Co. When these radionuclides, like $^{110m}$Ag and $^{125}$Sb, are not considered in the model, overestimates of up to 60% can result.

Scaling factor uncertainties are both random and systematic. The largest sources of uncertainty are attributable to potential materials composition variances. There also appear to be discrepancies in both the $^{94}$Nb and the $^{99}$Tc calculated variances considering NUREG/CR-3474 composition data as the basis for computing the potential uncertainty. The neutronics related uncertainties are not shown as random to illustrate that flux ratio variances simultaneously increase and decrease certain scaling factors. However, these uncertainties are random and the values shown represent maximums.

The potential to consistently overestimate $^{60}$Co content by about 25% minimum could lead to overestimates of the scaled radionuclide concentrations, and affect classification. This is offset by the potential for underestimating $^{94}$Nb, $^{99}$Tc, and $^{14}$C scaling factors considering NUREG/CR-3474 composition data. There is also an apparent discrepancy in both the $^{94}$Nb and $^{99}$Tc uncertainties, since the activation results are comparable to the conservative direct sample results.

Method uncertainty can be improved in several areas. Discrete component specific models will reduce the systematic uncertainty in $^{60}$Co determination. Proper consideration of source contents for components containing tin and silver will also reduce this uncertainty. Scaling factor uncertainties can be reduced by development of a comprehensive materials composition data base to supplement NUREG/CR-3474. These composition data will also help resolve the apparent discrepancies in the $^{59}$Ni, $^{94}$Nb, and $^{99}$Tc scaling factors.

## 6.0 RECOMMENDATIONS

The results of this study indicate there are uncertainties associated with the determination of Part 61 radionuclide concentrations in irradiated hardware for both characterization methods currently used. The sources of uncertainty and their effect on hardware classification vary with the method employed and the techniques used to implement the method. This section presents recommendations for improving current practices.

### 6.1 General Areas of Improvement

There are several areas where additional work will not only improve the accuracy of both methods, but will provide better comparisons of method results. These areas of improvement are discussed below.

#### 6.1.1 Relative Performance of Survey Instruments

Both methods rely on gamma radiation surveys of hardware components in spent fuel pool environs. They also rely on consistent application of survey procedures and practices. The uncertainty in the concentration of $^{60}$Co as determined through underwater gamma dose rate measurements may be minimized through relatively simple improvements in current measurement techniques. These improvements include:

1. Development of standard procedures for underwater surveys of hardware.
2. Determining the relative sensitivity of the different instruments employed as a function of source-detector distance.
3. Development of standard calibration procedures for underwater use of gamma probes.

The effort needed to make these improvements is iterative and should be performed in a field environment. The recommended effort would initially involve the development of under water survey procedures to be employed on a BWR and PWR hardware characterization project. These procedures would provide for detailed radiation profiling of pre-selected components having well known histories and materials compositions. Each component would be extensively profiled using three different manufacturers' instruments calibrated according to manufacturer's instructions. Radiation profiles would be obtained using a fixture which assures a fixed source-detector geometry. Axial readings would be taken at contact and non-contact distances ranging from 6 to

12 inches from the component. Contact profiles would also be taken across the flats of BWR CRB's and PWR RCCA's to determine the variances due to local hot spots.

Analysis of these survey results will provide information on the relative accuracy of the instruments and on the degree of uncertainty related to hot spots and source-detector distance. These efforts would also provide the basis for updating procedures and developing instrument calibration standards for underwater applications.

### 6.1.2 Development of Component Specific Characteristics

As illustrated in the component data presented in Section 4, there are variances in the values currently used for component volumes. Since the metal volume is a critical parameter in classification and most components are standard according to LWR type and vendor, a consistent set of component parameters should be defined and published for industry use.

### 6.1.3 Resolution of the $^{59}$Ni Discrepancy

As discussed in previous sections and analyzed in Appendix B, there is a method dependent difference in the reported concentrations for $^{59}$Ni. The review performed for this study indicates that the direct sample method overestimates $^{59}$Ni concentrations. Due to the importance of $^{59}$Ni, as illustrated in Table 4.3, this discrepancy should be resolved. Two approaches are available: (1) perform more extensive and varied radiochemical analyses for $^{59}$Ni on hardware samples, and (2) conduct more extensive investigations of the relative behavior of $^{59}$Ni relative to $^{63}$Ni using calculational techniques. Both approaches should be pursued.

### 6.1.4 Determination of $^{94}$Nb Scaling Factors

This radionuclide has a major effect on classification. The direct sample method typically obtains LLD values from the radiochemical analysis of bulk metal samples and thus overestimates the $^{94}$Nb concentrations by using LLD based scaling factors. The scaling factors used to date by application of the activation analysis methods are comparable to or higher than those reported for the direct sample method (See Section 4.2.2). Thus, these results also appear to be overconservative. Even though both methods overestimate the concentrations, more accurate $^{94}$Nb scaling factors should be developed. This is consistent with the industry objective of not using LLD's as the basis for determining Part 61 radionuclide scaling factors. More accurate scaling factors can be determined through the use of improved radiochemical analyses techniques and the development of a more extensive niobium materials composition data base. The current radiochemical techniques could possibly be modified to permit the analysis of higher specific activity samples or the sensitivity of detecting $^{94}$Nb relative to $^{60}$Co could be improved. Preparation of a component materials data base which would address niobium is described below.

## 6.2 Improvements Specific to the Direct Sample Method

Areas where the accuracy of the direct sample method can be improved relate primarily to the practices employed during a typical project. These areas are discussed below.

### 6.2.1 Representativeness of Samples

It is not clear that all components in a given batch have irradiation and cooling times similar to the representative component selected for direct sampling. In addition to batching components of like constituents, batching components according to like irradiation history and cooling time could essentially eliminate this source of uncertainty.

### 6.2.2 Differences in Radiation Profiling Conditions

The use of the same profiling procedures, personnel, and dose rate instrumentation would reduce the potential uncertainties resulting from component characterization in two distinct phases by different organizations.

## 6.3 Improvements Specific to the Activation Analysis Methods

This method is currently used by vendors, government contractors and electric utilities. Some areas of improvement relate to the overall method while others relate to particular techniques. Generally applicable areas for improvement are discussed below.

### 6.3.1 Materials Composition Data Base

All activation analysis techniques rely on constituent compositions. Some techniques, like those employed by WMG and TPT use these data on a relative basis to determine scaling factors. Other techniques, like those employed by PNL and some utilities, use these data to calculate base metal activation. A statistically valid materials composition data base which addresses the range of compositions in hardware metals for key Part 61 radionuclide precursors would lead to consistent use of compositions. This would reduce the uncertainties now present due to the use of varying initial materials compositions. It would also facilitate comparison of characterization methods by narrowing the range of variables involved in activation analysis. Additionally, it could provide a narrower base for determining $^{94}$Nb scaling factors, as described in Section 6.1.4 above.

### 6.3.2 Scaling Factor Data Base

Scaling factor determination is a critical factor in classification. The data published in this report is the first attempt to present information on the ranges of expected scaling factors by component type and other factors. A component specific scaling factor data base which relies on both direct sample data and activation analysis results would provide general guidance to all hardware generators and disposal sites. It could also provide data to eliminate LLD based scaling factors.

### 6.3.3 $^{60}$Co Dose-to-curie Conversion Factors

Activation analysis techniques rely on computer programs to calculate component specific $^{60}$Co dose-to-curie conversion factors. The models used to depict component geometry vary from relative simple cylinders (LPRM's) and flat plates (poison curtains), to fairly complex composites (control rod blades and cluster assemblies). The analytical techniques used by the different programs, and the simplifying assumptions employed to model different component geometries, should be evaluated on an absolute and relative basis. This evaluation should also address the effects of gamma backscatter. The results would provide a better understanding of the relative accuracy of the different codes and reduce the uncertainty in $^{60}$Co determination.

## 7.0 REFERENCES

1.* "Nuclide Characterization of Fuel Pool Stored Components - Pilgrim Nuclear Power Station - Final Report", Science Applications International Corporation, SAIC-85/1696, May 3, 1985.

2. "RADCOR - A Computer Code to Characterize, Classify and Document Packaged Irradiated Hardware", (Design Report) WMG, Inc., Mohegan Lake, New York, WMG 8502 Revision 1, October 1985.

3. "MICROSHIELD User's Manual Version 2.0", Grove Engineering, Washington Grove, Maryland, 1985.

4. "RADMAN - A Computer Code to Classify And Document Packaged LLW in Accordance With 10 CFR Part 61 Regulations", (Topical Report) Waste Management Group, Inc., New York, WMG-102-NP-A Volumes 1, 2, and 3, May 1983.

5. Su, S., Baylor, K.J., and Engholm, B.A., "PATH Gamma Shielding Code User's Manual", G.A. Technologies, GA-A16772, September 1982.

6. Walti, P. and Koch, P., "MICROX - A Two Region Flux Spectrum Code For Efficient Calculations of Group Cross Sections", Gulf General Atomic Company, Gulf-GA-A10827, April 1972.

7. NUREG/CR-3474, "Long-Lived Activation Products in Reactor Materials", Evans, J.C., Pacific Northwest Laboratory, August 1984.

8. "10 CFR 61 Characterization Using Direct Assay Technology at Vermont Yankee", Battelle Pacific Northwest Laboratories, August 1985.

9. ORNL/TM-7175, "A User's Manual For The ORIGEN2 Computer Code", Croff, A.G., Oak Ridge National Laboratory, July 1980.

10. "SLAB", A Battelle gamma shielding computer code. "Proprietary information. Not publicly available."

11. "Repository Disposal Requirements For Spent Fuel Disassembly Hardware and Other Non-Fuel Components", (Letter Report to DOE) Pacific Northwest Laboratory, July 1985.

12. "LOTUS 1-2-3 User's Manual", Lotus Development Corporation, Cambridge, Massachusetts, 1983.

13.*"Waste Characterization Technique for Fuel-Pool Stored Components", Science Applications International Corporation, SAIC-87/1054 Revision 1, April 1987.

14. Telecon between Dr. Seymore J. Rafferty (Dairyland Power) and A. Levin (WMG, Inc.).

15. Technical Manual For Eberline Model RO-7, Eberline Instrument Corporation, November 1985.

16. NUREG/CR-4101, "Assay of Long-Lived Radionuclides in Low-Level Wastes From Power Reactors", Cline, J.E., et. al., Science Applications International Corporation, February 1985.

17. Engle, R.L., et. al., "ISOSHLD - A Computer Code For General Purpose Isotopic Shielding Analysis", Richland, Washington, BNWL-2316, June 1966.

18. "QAD-CGGP - A Combinatorial Geometry Version Of WAD-P5A, A Point Kernel Code System For Neutron And Gamma-Ray Shielding Calculations Using The GP Buildup Factors", ORNL, April 1986.

19. Technical Associates Specification Data Sheet For The CP-MU High Range Underwater Monitor Model CP-MU, CP-MU-1000 and Model CP-MU-7, Technical Associates.

20. Granados, B.L., et. al., "Evaluation Of Dositec Instruments In PWR Spent Fuel Pool", Dositec, Inc., Framingham, Massachusetts, 1986.

21. Dositec Specification Data Sheet For the Model PR-2 & PR-7 Portable Remote Monitor, Dositec, Inc.

22. *Annual Book of ASTM Standards, 1986*, American Society for Testing Materials, Philadelphia, 1986.

23. Stevens, J.J., Jr., and Pohl, R.D., "Trace Elements in Reactor Steels: Implications for Decommissioning", Laboratory of Atomic and Solid State Physics and Material Science Center, Cornell University, Report 2882, 1977.

24. Peckner, D. and Bernstein, I., *Handbook of Stainless Steels*, McGraw-Hill Book Company, Inc., New York, 1977.

25.*"Nuclide Characterization of Fuel Pool Stored Components - Dresden's Unit 2 and Unit 3 Fuel Pools - Final Report", Science Applications International Corporation, SAIC-85/1737, June 10, 1985.

26.*"Nuclide Characterization of Fuel Pool Stored Components - Brown's Ferry Atomic Power Station Unit 1 - Final Report", Science Applications International Corporation, SAIC-85/1633, April 1985.

27.*"Nuclide Characterization of Fuel Pool Stored Components - Nine Mile Point Unit 1 Fuel Pool - Final Report", Science Applications International Corporation, SAIC-86/1915, October 21, 1986.

28.*"Methodology for Radiation Measurements and Analysis of Spent Control Rods at Oyster Creek Nuclear Generating Station", Torrey Pines Technology, TDR-541, June 1984.

29.*"Control Rod Blade Packaging Plan for Oyster Creek - Final Report", Waste Management Group, Inc., WMG-8507, July 22, 1985.

30.*"H.B. Robinson 2 Plant Spent Fuel Pool Cleanup Project Report", Waste Management Group, Inc., WMG-8601, January 1986.

31.*"Sequoyah Station Spent Fuel Pool Cleanup Project Report", Waste Management Group, Inc., WMG-8607, July 1986.

32.*"Preliminary Packaging Plan for Irradiated Hardware at the Salem Station", Waste Management Group, Inc., WMG-8608, October 1986.

33.*"Nuclide Characterization of Fuel Pool Stored Components - Brunswick Unit 1 Fuel Pool - Final Report", Science Applications International Corporation, SAIC-86/1122 Revision 1, July 11, 1986.

* An asterisk indicates reports used to prepare Section 4.0 data base.

# APPENDIX A

# CHARACTERIZATION AND CLASSIFICATION METHODOLOGIES

## A.1 SAIC METHODOLOGY

The SAIC characterization and classification methodology for irradiated hardware is a combination of direct sampling and analytical techniques. This discussion references documents found in Section 7.

Estimates of $^{60}$Co concentrations throughout a component are made through the direct measurement of the dose rate-to-curie conversion coefficient for $^{60}$Co at a specific location on the component. The relative concentrations of hard-to-measure Part 61 radionuclides are calculated from scaling factors developed from direct sample data analyses.

Relative concentrations of activation products in irradiated hardware depend upon the chemical constituents in the base material and upon the neutron exposure history of the material. Elemental concentrations are generally not well known and the neutronics history is often difficult to obtain. Additionally, some of the chemical constituents which influence Part 61 classification are activated trace contaminants. Sampling and radiochemical analysis theoretically eliminates much of this uncertainty and allows for more accurate and realistic results.

### A.1.1 Estimation of Cobalt-60 Concentration

The estimation of $^{60}$Co concentrations in irradiated hardware is performed in two steps. A quantitative determination of $^{60}$Co is made using the SAIC Quantiscan (tm) system. The gamma scan determines the concentration of $^{60}$Co at the scan location. Following the gamma scan, dose rate measurements are made at the gamma scan location and along the length of the component, where applicable, to measure the variation in the concentration of $^{60}$Co along the length of the component.

#### A.1.1.1 Gamma Scanning Technique

SAIC uses the Quantiscan (tm) system to perform a quantitative determination of $^{60}$Co concentrations in irradiated hardware. The Quantiscan (tm) system consists of a tungsten shielded, nitrogen-cooled, lithium-drifted germanium (GeLi) detector, and a computer-based data acquisition and analysis system. The detector is mounted on a cart and an air collimator is affixed opposite the detector surface. The collimator eliminates attenuation of gamma radiation by water. To perform gamma scans, the detector is aimed by lowering the collimator into the spent fuel pool and placing it up against the irradiated component.

Prior to use, the Quantiscan (tm) system is calibrated using a certified mixed-isotope point source standard to measure both on-axis and off-axis detector response for the entire gamma energy spectrum. Scan measurements are made at several distances, both on- and off-axis, and are curve fitted to obtain values for parameters describing detector response. Using these parameters, previously developed equations are solved for a particular hardware geometry to calculate detector efficiency for that geometry.

Once calibrated in this manner, the Quantiscan (tm) system can measure the curie concentration of $^{60}$Co in irradiated hardware in microcuries per cubic centimeter. Counting data is collected by the computer data acquisition system and is analyzed using a three-stage analysis process. First, a computer program is run to calibrate each channel of the multi-channel analyzer to the energy spectrum data of a $^{228}$Th check source. Second, a background radiation spectrum is analyzed to enable the background radiation (due almost entirely to pool water $^{60}$Co contamination) to be subtracted from the scan data.

Finally, the computer analysis program uses internally stored algorithms to locate spectral data peaks and identify isotopes by comparing the peaks to the data in a radionuclide library. Once identified, integration routines are used to sum the count rate data and to determine the $^{60}$Co concentrations in uCi/cm$^3$.

The accuracy of scanning underwater fuel-pool components depends upon the accuracy of the component dimensions, the distance between the end of the Quantiscan (tm) collimator and the component, the orientation of the component relative to the collimator end, the alignment of the long collimator, and the uniformity of the distribution of activities throughout the component.

### A.1.1.2 Dose Rate Profiling

After the $^{60}$Co concentration has been measured using the Quantiscan (tm) system, a dose rate-to-curie conversion factor is calculated by measuring the contact dose rate at the gamma scan location and comparing it to the curie content of $^{60}$Co at that location.

Dose rate measurements are made using an RO-7 Ionization Chamber. Subsequent contact dose rate measurements are made at measured (approximately 1 foot) increments along the longitudinal axis of each component, where appropriate. At each measurement location, the survey meter is positioned to minimize the source-detector distance.

The dose rate profile is later used to scale the concentrations of all Part 61 radionuclides along the longitudinal axis of the component. This results in a profile of curies per unit volume and the total integrated curie content of the component.

### A.1.2 Estimation of Part 61 Scaling Factors

Concentrations of hard-to-measure Part 61 radionuclides are estimated via direct sampling/radiochemistry techniques. The SAIC Fuel Pool Sampler is capable of in-situ sampling of various irradiated hardware in the spent fuel pool. This device physically grinds small samples (typically between 0.01 and 0.3 grams) of material from the surface of the component at the location of the gamma scan. The particles are collected in an underwater filter and retrieved for radiochemical analysis.

Radiochemistry and radiometric techniques are employed to calculate the radionuclide scaling factors for Part 61 and other detectable radionuclides, relative to $^{60}$Co or $^{137}$Cs. Activated corrosion products and $^{137}$Cs are scaled to $^{60}$Co. Fission products of transuranics are subsequently scaled to $^{137}$Cs. Distillation and separation chemistry techniques are used to isolate radioisotopic samples, which are in turn measured by appropriate counting equipment.

#### A.1.2.1 Gross Gamma Measurements

Prior to any radiochemical analysis, each sample is gamma-spectrum analyzed using a GeLi gamma ray spectrometer. The spectrometer computer system automatically searches the acquired pulse-height spectrum for peaks; calculates the energy, the gross and net (sample minus background) counts; and identifies the radionuclides giving rise to each peak. The data is also corrected for radioactive decay from the sampling time. From this data, the relative concentrations of the various gamma-emitting isotopes in the sample can be measured in microcuries per unit weight (of the sample).

The gross gamma measurements yield the relative concentrations of the scaling isotopes, $^{60}$Co and $^{137}$Cs, in the sample. These relative concentrations will be used to scale other radionuclides as they are chemically extracted from the original sample.

#### A.1.2.2 Low Energy Beta-Emitting Radionuclides

The low energy beta-emitting radionuclides ($^{3}$H, $^{14}$C, $^{63}$Ni, $^{241}$Pu) are separated from the original sample through liquid extraction processes, remaining in solution. The liquid aliquots of the radiochemically processed samples are counted using a liquid scintillation counter. The relative concentration of each radionuclide is measured in terms of microcuries per unit weight of the original sample and decay corrected to the time of sampling.

### A.1.2.3 High Energy Beta-Emitting Radionuclides

The high energy beta-emitting radionuclides ($^{89}$Sr, $^{90}$Sr, $^{99}$Tc) are easily counted using a thin-window gas-flow proportional counter, after being separated from the original sample. The relative concentration of each radionuclide is measured in microcuries per unit weight of the original sample and decay corrected to the time of sampling.

### A.1.2.4 X-Ray-Emitting Radionuclides

Radionuclides emitting with low energy photon emission ($^{55}$Fe, $^{59}$Ni, $^{129}$I) are chemically separated and counted using NaI(Tl) spectrometers. Relative concentrations are measured in microcuries per unit weight of the original sample and decay corrected to the time of sampling. For activated hardware, SAIC no longer analyzes for $^{59}$Ni. Instead, scaling factor data from NUREG/CR-4101 is used.

### A.1.2.5 Gamma-Emitting Radionuclides

Relative concentrations of gamma-emitting radionuclides ($^{94}$Nb, $^{144}$Ce) which cannot readily be measured in the gross gamma scan, are chemically extracted from a portion of the original sample. Once extracted, a GeLi gamma spectrometer is used to measure the relative concentrations in microcuries per unit weight of the original sample. The results are decay corrected to the time of sampling.

### A.1.2.6 Alpha-Emitting Radionuclides

Uranium and most transuranics emit alpha particles which are detected using silicon surface barrier pulse-height spectrometers. SAIC utilizes these detectors to count the gross alpha activity of the U, Pu, Am, and Cm radionuclides. Liquid extraction techniques separate the alpha emitting radionuclides into three groups consisting of U, Pu, and Am/Cm. Counts are corrected for radionuclide decay to the time of sampling.

### A.1.2.7 Scaling Factor Development

Data collected from the various chemical extractions and counting systems are used to develop scaling factors. Activation products ($^{14}$C, $^{55}$Fe, $^{59}$Ni, $^{63}$Ni, $^{94}$Nb) in the component samples are scaled to the relative concentration of $^{60}$Co in the sample.

Scaling factors for fission products ($^{90}$Sr, $^{99}$Tc, and $^{129}$I) are determined from a scraping sample taken from the surface of the component, and scaled to the relative concentration of $^{137}$Cs in the sample. Similarly, scaling factors for the transuranics ($^{241}$Pu, Cm-242 and total TRU's) are determined from the scraping sample and scaled to the relative concentration of $^{137}$Cs.

### A.1.3 Estimation of Radionuclide Concentrations

Radionuclide concentrations for Part 61 radionuclides are calculated from the gamma scan measured concentrations of $^{60}$Co and the appropriate scaling factors, for the location of the direct sample. The dose rate-to-curie conversion factor is used with the dose rate measurements away from the direct sample location to estimate the $^{60}$Co concentrations at the measured dose rate locations. Again, the scaling factors are used to estimate the concentration of the hard-to-measure radionuclides.

#### A.1.3.1 Activation Product Concentrations

Component concentrations of the activation products ($^{14}$C, $^{55}$Fe, $^{59}$Ni, $^{63}$Ni and $^{94}$Nb) are scaled to $^{60}$Co by comparing the relative concentration of each of the radionuclides to the relative concentration of $^{60}$Co in the direct sample from the radiochemistry data. The concentrations are found by applying the scaling factors to the Quantiscan (tm) measured $^{60}$Co concentration at the gamma scan location, and to the $^{60}$Co concentrations determined by measuring the dose rate at other axial locations and multiplying by the dose-to-curie conversion factor.

#### A.1.3.2 Fission Product Concentrations

Determination of tramp fission product concentrations on the surface of the component is complex. SAIC uses its sampler to obtain a scraping of material on the component surface. If $^{137}$Cs exists in this direct sample, it indicates a buildup of fission products on the surface of the component. To properly account for this tramp fission product accumulation, a surface-to-volume ratio must be calculated for the component. This ratio is then applied to the $^{137}$Cs/$^{60}$Co scaling factor determined by the radiochemical analysis of the sample. The correction accounts for the fact that the tramp fission products represent a very thin layer on the surface of the component.

A radiochemical analysis of the scraping sample provides the scaling factors of fission products relative to $^{137}$Cs. The concentrations of the various tramp fission products are calculated by applying the adjusted $^{137}$Cs/$^{60}$Co scaling factor to the tramp fission product/$^{137}$Cs scaling factor determined by the radiochemistry data, and applying the overall factor to the $^{60}$Co concentrations as is done with the activation products.

### A.1.3.3 Transuranic Concentrations

Transuranic concentrations are determined in a manner similar to the determination of tramp fission product concentrations. The concentrations of the various transuranics are calculated by applying the adjusted $^{137}Cs/^{60}Co$ scaling factor to the transuranic/$^{137}Cs$ scaling factor determined by the radiochemistry data, and applying the overall factor to the $^{60}Co$ concentrations as with the activation products.

## A.2 WMG RADCOR

The WMG characterization and classification methodology for irradiated hardware uses activation analysis techniques to estimate radionuclide concentrations. Estimates of the $^{60}$Co concentrations are made, from which the concentrations of other radionuclides are scaled.

This methodology requires that the chemical constituents of the hardware and the irradiation history be well known. Additionally, accurate underwater radiation measurements are needed to estimate $^{60}$Co concentrations. This discussion references documents found in Section 7.

### A.2.1 Estimation of Cobalt 60 Concentration

Estimation of $^{60}$Co concentrations in irradiated hardware is performed in two steps. The first step is to develop dose rate-to-curie conversion factors for the component geometry. This is performed by using a point kernel gamma shielding code, which for WMG is the MICROSHIELD computer code.

The irradiated hardware (source) is modeled for input into MICROSHIELD. This may include some homogenization of the source geometry to fit the available geometry models given in the code. For example, a PWR Thimble Plug will be homogenized into a cylindrical source.

The source is then given a known $^{60}$Co content, generally one curie for the whole source. A gamma dose rate is calculated at distances underwater (usually 6 and 12 inches) which will later be used in the in-situ underwater gamma dose rate measurement of the component. Based upon the gamma dose rate calculated, a dose rate-to-curie conversion factor is calculated for the component geometry of interest.

Component characterization may employ more than one dose rate-to-curie conversion factor along its axial length. For instance, a BWR Control Rod Blade has a varying geometry along its axial length. The section of the blade near the handle will have one dose rate-to-curie conversion factor, the cruciform section will have another, and the velocity limiter section yet a third. RADCOR is designed to accommodate this component variation.

The WMG RADCOR computer code, which performs the characterization and classification processes, contains previously calculated dose rate-to-curie conversion factors for several PWR and BWR components at distances of 6 inches. The code allows for the manual entry of dose rate-to-curie conversion factors at distances other than 6 inches, when desired.

The second step in the $^{60}$Co concentration estimate is the input of the measured dose rates along the component axis into the RADCOR code. Using either the internal dose rate-to-curie or the user-input values, RADCOR calculates the $^{60}$Co concentration at the measurement location for the component of interest.

### A.2.2 Estimation of Part 61 Scaling Factors

Internally, the RADCOR computer code performs an "reverse activation analysis" to calculate the Part 61 scaling factors. To perform this task, the code needs the following input.

1. A component specific data base. For each component type (i.e., PWR Thimble Plug, BWR Control Rod Blade, etc.), a data base of information must be collected. These data will include the component chemical composition, the component irradiation and decay history, and the neutron fluxes seen by the components in the core.

2. The dose rate-to-curie conversion factors as calculated by the MICROSHIELD computer code.

3. For the larger components, underwater radiation profiles for each component along its axial length at the distance for which the dose rate-to-curie conversion factor was calculated. Smaller components require a single dose rate measurement. These dose rate measurements are made using an RO-7 Ionization Chamber.

With this information, the RADCOR code begins by performing a calculation to estimate the $^{60}$Co content in the component at the time of the radiation measurement. This $^{60}$Co content is based upon the radiation measurement(s) made and the internally or externally provided dose rate-to-curie conversion factors.

Next, RADCOR decay corrects the $^{60}$Co content back to the time of component removal from the reactor neutron flux, a date provided by the user. Then, utilizing the thermal and fast neutron fluxes (a two-group calculation), the neutron cross sections, and the radionuclide half-lives, RADCOR calculates a radionuclide scaling factor for each of the radionuclides of interest relative to the activation of $^{60}$Co. Since RADCOR assumes a homogenized component material distribution, the activities of $^{3}$H, $^{110m}$Ag and other radionuclides created within the component (CPB's and RCCA's, for example) are properly calculated.

### A.2.3 Estimation of Radionuclide Concentrations

The estimation of radionuclide concentrations is performed by RADCOR utilizing the radionuclide scaling factors at the time of component removal from the reactor neutron flux. RADCOR will estimate the radionuclide concentrations and subsequently classify the component in accordance with Part 61.

Using the measured dose rates (at up to nine axial locations for the larger components) and the dose rate-to-curie conversion factors for the components, the $^{60}$Co concentrations are first estimated at various locations. Then, using the scaling factors calculated, the concentrations of other Part 61 Table 1 and Table 2 radionuclides (excluding transuranics and fission products) are calculated. Finally, RADCOR decay corrects the radionuclide concentrations to the date of characterization and volume averages the concentrations.

Transuranics are considered as being present on components as surface contaminants. To address transuranic radionuclide concentrations, RADCOR assumes that the wetted surface area of each component is covered with a two mil uniform layer of spent fuel pool silt. Representative fuel pool silt sample data may be input to RADCOR, and used to estimate the transuranic content, as a surface contaminant, for each component.

## A.3 TORREY PINES TECHNOLOGY

The methodology used by Torrey Pines to characterize and classify irradiated hardware is very similar to the WMG methodology with some subtle differences. The differences between the methodologies are discussed below. This discussion references documents found in Section 7.

### A.3.1 Estimation of Cobalt-60 Concentrations

Similar to the WMG methodology, Torrey Pines estimates the $^{60}$Co concentrations in irradiated hardware through the use of dose rate-to-curie conversion factors generated by a point-kernel shielding computer code. In Torrey Pines' methodology, the PATH computer code is used to generate the dose rate-to-curie conversion factors. Similar to the MICROSHIELD code used by WMG, the PATH code employs a single radioactive source, requiring some simplification of source geometries to "fit" the available models in the code.

### A.3.2 Estimation of Part 61 Scaling Factors

TPT does not have available a computer code to perform the "reverse activation analysis" similar to WMG's RADCOR code. Instead, scaling factors are calculated by hand. The methodology is the same as that employed by the RADCOR code, with the exception of the generation of neutron cross sections. TPT uses the MICROX two-region spectrum code to calculate two-group cross sections used in the activation analysis calculations.

### A.3.3 Estimation of Radionuclide Concentrations

The estimation of radionuclide concentrations as performed by the TPT methodology is the same as the WMG methodology used in RADCOR. The scaling factors are applied to the estimated concentration of $^{60}$Co, and concentrations of other radionuclides are calculated.

Literature on the TPT methodology does not discuss a methodology for calculating tramp transuranic concentrations.

## A.4 BATTELLE PACIFIC NORTHWEST LABORATORIES

Battelle Pacific Northwest Laboratories (PNL) employs a characterization methodology which is very similar to the WMG and TPT methodologies. However, the manner in which scaling factors are obtained is somewhat different. This section references documents in Section 7.

In NUREG/CR-3474, PNL used an in-house activation analysis computer program to calculate the activation of various PWR and BWR reactor internals over the anticipated life of the plant (taken to be 40 years). The material compositions used in the analyses were experimental data generated for the purposes of the NUREG study. PNL uses these calculations as the source for scaling factors for irradiated hardware.

### A.4.1 Estimation of Cobalt-60 Concentrations

Estimates of the $^{60}$Co concentrations in irradiated hardware using the PNL methodology are very similar to the WMG and TPT methodologies. Dose rate-to-curie conversion factors are developed for the component geometry using the SLAB gamma shielding computer code. The SLAB code is somewhat superior to the MICROSHIELD and PATH codes in that it allows calculation of multiple sources. This factor allows for a more exact representation of the source geometry.

The available literature does not discuss whether more than one dose rate-to-curie conversion factor is calculated for large components. Additionally, the literature does not discuss the distance at which the dose rate-to-curie calculations are made.

As with the WMG and TPT methodologies, dose rates are measured for the various components, and using the dose rate-to-curie conversion factors, a $^{60}$Co concentration is estimated. However, unlike the WMG and TPT methodologies, ionization chambers are not used to measure the gamma dose rates. The PNL methodology instead relies upon thermoluminescent dosimeters (TLD's) to measure the gamma dose rates.

### A.4.2 Estimation of Part 61 Scaling Factors

Scaling factors for the Part 61 radionuclides are chosen from the calculations performed for the NUREG study, and from other databases. They are calculated by comparing the $^{60}$Co concentration to the concentration of the radionuclide of interest, at the appropriate irradiation history. No credit is taken for radionuclide decay in this calculation which will produce conservative scaling factors for radionuclides with half-lives greater than $^{60}$Co (most of the Part

61 radionuclides), and non-conservative scaling factors for those radionuclides with half-lives less than $^{60}$Co (for instance, $^{55}$Fe).

### A.4.3 Estimation of Radionuclide Concentrations

As shown in 4.2, scaling factors for Part 61 radionuclides are generated from the calculated radionuclide concentrations.

To estimate the transuranic concentrations of fission chambers, PNL utilizes the ORIGEN-2 computer code to perform an activation analysis of fuel material. In this calculation, fuel material of the isotopic composition of the material in a fission chamber is used as input. Output from the calculation provides a direct estimation of the various transuranics in the chamber.

The available literature does not discuss the techniques used to estimate surface contamination of irradiated components.

# APPENDIX B

# ANALYSIS OF $^{59}$NI/$^{63}$NI RELATIONSHIP

## B.1 INTRODUCTION

The review of available data indicates that the direct sample method consistently uses a $^{59}$Ni/$^{63}$Ni relationship that is higher than that used by activation analysis methods by a factor of 2 or more. This discrepancy is explored and discussed in this Appendix.

## B.2 EVALUATION USING NUREG/CR-3474 METHOD

The first approach taken to evaluate this phenomenon was to calculate the ratio of $^{59}$Ni to $^{63}$Ni which should be relatively constant. This consistency is due to the following:

1. The constant ratio, approximately 19, of $^{58}$Ni/$^{62}$Ni atoms in the unirradiated metal.
2. The relatively long half lives of the radioactive species (100 years for $^{63}$Ni and 80,000 years for $^{59}$Ni).
3. Similar ratios of the thermal-to-epithermal neutron activation cross sections (approximately 30).
4. The activation of the precursor radionuclides is independent of the fast-to-thermal neutron flux ratios.

To further investigate the $^{59}$Ni/$^{63}$Ni relationship, a computer run was made using the PNL methodology for performing activation analyses as given in NUREG/CR-3474. The analysis was performed using the Type 304L stainless steel composition data, and the Westinghouse PWR core shroud flux data as given in the NUREG document. Radionuclide libraries were assembled and reviewed for accuracy for this study. The output from this calculation for the two radionuclides of nickel is attached as Figures B.1 and B.2.

The PNL methodology computes the $^{59}$Ni/$^{63}$Ni ratio to be between 0.007 and 0.008. This ratio exists regardless of the neutron exposure and within the range of one to ten years decay after core removal. The SAIC data described in Section 4, however, show the ratio of $^{59}$Ni/$^{63}$Ni to vary between 0.01 and 0.12. This data variance does not conform to the expected behavior of the $^{59}$Ni/$^{63}$Ni relationship.

## Figure B.1 NUREG/CR-3474 Method Output For Ni-59

```
NICKEL       59          HALF-LIFE= 8.00E+04 YEARS

DIRECT CAPTURE TARGET ELEMENT CONCENTRATION =100000.00 PPM

N-2N REACTION INCLUDED

WESTINGHOUSE PWR SHROUD VOLUME AVERAGED FLUX

                                            BURNUP RATIOS FOR Z=28
T(YEARS)            ACTIVITY(CI/GM)       58        59        60
      1.                   5.41E-06    9.989    ******     9.994
      2.                   1.07E-05    9.979    ******     9.988
      3.                   1.58E-05    9.968    ******     9.983
      4.                   2.07E-05    9.958    ******     9.978
      5.                   2.55E-05    9.947    ******     9.974
      6.                   3.01E-05    9.937    ******     9.971
      7.                   3.46E-05    9.926    ******     9.969
      8.                   3.90E-05    9.916    ******     9.967
      9.                   4.33E-05    9.905    ******     9.965
     10.                   4.74E-05    9.895    ******     9.964
     11.                   5.13E-05    9.885    ******     9.964
     12.                   5.52E-05    9.874    ******     9.964
     13.                   5.90E-05    9.864    ******     9.964
     14.                   6.26E-05    9.853    ******     9.965
     15.                   6.61E-05    9.843    ******     9.966
     16.                   6.95E-05    9.833    ******     9.968
     17.                   7.28E-05    9.822    ******     9.970
     18.                   7.60E-05    9.812    ******     9.973
     19.                   7.92E-05    9.802    ******     9.976
     20.                   8.22E-05    9.791    ******     9.980
     21.                   8.51E-05    9.781    ******     9.983
     22.                   8.79E-05    9.770    ******     9.987
     23.                   9.07E-05    9.760    ******     9.992
     24.                   9.33E-05    9.750    ******     9.997
     25.                   9.59E-05    9.740    ******    10.002
     26.                   9.84E-05    9.729    ******    10.007
     27.                   1.01E-04    9.719    ******    10.013
     28.                   1.03E-04    9.709    ******    10.019
     29.                   1.05E-04    9.699    ******    10.025
     30.                   1.08E-04    9.688    ******    10.032

   ACTIVITY(CI/GM)          YEARS AFTER SHUTDOWN
         1.08E-04                      1.00E+00
         1.08E-04                      1.00E+01
         1.08E-04                      1.00E+02
         1.07E-04                      1.00E+03
         9.78E-05                      1.00E+04
         4.11E-05                      1.00E+05
         7.09E-09                      1.00E+06
```

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**Figure B.2** NUREG/CR-3474 Method Output For Ni-63

```
NICKEL       63          HALF-LIFE= 1.00E+02 YEARS

DIRECT CAPTURE TARGET ELEMENT CONCENTRATION =100000.00 PPM

N-GAMMA AND FISSION REACTIONS ONLY

WESTINGHOUSE PWR SHROUD VOLUME AVERAGED FLUX

                                          BURNUP RATIOS FOR Z=28
T(YEARS)           ACTIVITY(CI/GM)      56       59        60       61        62       63
      1.                 7.39E-04    9.989   ******     9.994   10.154     9.968   ******
      2.                 1.47E-03    9.979   ******     9.988   10.308     9.936   ******
      3.                 2.18E-03    9.968   ******     9.983   10.462     9.904   ******
      4.                 2.89E-03    9.958   ******     9.978   10.615     9.873   ******
      5.                 3.58E-03    9.947   ******     9.974   10.768     9.841   ******
      6.                 4.27E-03    9.937   ******     9.971   10.922     9.810   ******
      7.                 4.94E-03    9.926   ******     9.969   11.075     9.779   ******
      8.                 5.60E-03    9.916   ******     9.967   11.228     9.748   ******
      9.                 6.25E-03    9.905   ******     9.965   11.380     9.717   ******
     10.                 6.89E-03    9.895   ******     9.964   11.533     9.686   ******
     11.                 7.52E-03    9.885   ******     9.964   11.686     9.655   ******
     12.                 8.14E-03    9.874   ******     9.964   11.838     9.625   ******
     13.                 8.75E-03    9.864   ******     9.964   11.991     9.594   ******
     14.                 9.36E-03    9.853   ******     9.965   12.143     9.564   ******
     15.                 9.95E-03    9.843   ******     9.966   12.295     9.534   ******
     16.                 1.05E-02    9.833   ******     9.968   12.448     9.504   ******
     17.                 1.11E-02    9.822   ******     9.970   12.600     9.474   ******
     18.                 1.17E-02    9.812   ******     9.973   12.752     9.444   ******
     19.                 1.22E-02    9.802   ******     9.976   12.904     9.415   ******
     20.                 1.28E-02    9.791   ******     9.980   13.056     9.385   ******
     21.                 1.33E-02    9.781   ******     9.983   13.208     9.356   ******
     22.                 1.38E-02    9.770   ******     9.987   13.360     9.327   ******
     23.                 1.43E-02    9.760   ******     9.992   13.512     9.297   ******
     24.                 1.49E-02    9.750   ******     9.997   13.664     9.268   ******
     25.                 1.54E-02    9.740   ******    10.002   13.815     9.240   ******
     26.                 1.59E-02    9.729   ******    10.007   13.967     9.211   ******
     27.                 1.63E-02    9.719   ******    10.013   14.119     9.182   ******
     28.                 1.68E-02    9.709   ******    10.019   14.271     9.154   ******
     29.                 1.73E-02    9.699   ******    10.025   14.423     9.125   ******
     30.                 1.77E-02    9.688   ******    10.032   14.575     9.097   ******

   ACTIVITY(CI/GM)        YEARS AFTER SHUTDOWN
         1.76E-02                     1.00E+00
         1.64E-02                     1.00E+01
         8.22E-03                     1.00E+02
         8.03E-06                     1.00E+03
         6.33E-36                     1.00E+04
          .00E+00                     1.00E+05
          .00E+00                     1.00E+06
```

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## B.3 EVALUATION CONSIDERING SCALING FACTOR RANGES

To further investigate $^{59}Ni$ scaling factors, a series of parametric cases were run to define the expected range of scaling factors for all Part 61 radionuclides. The scaling factors for $^{14}C$, $^{55}Fe$, $^{59}Ni$, $^{63}Ni$, $^{94}Nb$ and $^{99}Tc$ were calculated using the PNL/NUREG methodology. Scaling factor ranges were calculated based upon the following assumptions:

1. The minimum scaling factor is based on an assumed one-year irradiation time with no radionuclide decay. For $^{55}Fe$ only, the minimum scaling factor is based on an assumed ten-year irradiation time with a ten-year radionuclide decay.

2. The maximum scaling factor is based on an assumed ten-year irradiation time with a ten-year radionuclide decay. For $^{55}Fe$ only, the maximum scaling factor is based on an assumed one-year irradiation time with no radionuclide decay.

3. The scaling factor range bandwidth is increased by an additional factor of two (total factor of 2) to account for the variability in the cobalt pre-irradiation concentration in the base metal.

4. The $^{94}Nb$ scaling factor range bandwidth is increased by an additional factor of two (total factor of 4) to account for the variability in the niobium pre-irradiation concentration in the base metal.

5. The $^{99}Tc$ scaling factor range bandwidth is increased by an additional factor of two (total factor of 4) to account for the variability in the molybdenum pre-irradiation concentration in the base metal.

6. The $^{14}C$ scaling factor range bandwidth is increased by an additional factor of two (total factor of 4) to account for the variability in the nitrogen pre-irradiation concentration in the base metal.

The resultant scaling factor ranges are presented in Table B.1 below.

Table B.1 Calculated Scaling Factor Ranges

| Isotope | Minimum Factor | Maximum Factor |
|---|---|---|
| $^{14}C$ | 1.03E-5 | 1.20E-3 |
| $^{55}Fe$ | 8.65E-1 | 1.88E+1 |
| $^{59}Ni$ | 1.34E-4 | 3.42E-3 |
| $^{63}Ni$ | 1.83E-2 | 4.97E-1 |
| $^{94}Ni$ | 2.04E-7 | 2.26E-5 |
| $^{99}Tc$ | 4.10E-7 | 4.11E-5 |

The reported scaling factors for $^{14}C$, $^{55}Fe$, $^{63}Ni$, $^{94}Nb$ and $^{99}Tc$ generally fall within the ranges of scaling factors presented above The reported scaling factors for $^{59}Ni$ as determined by the different activation analysis methodologies (WMG, TPT and PNL) also agree well with the tabulated range. However, the reported SAIC $^{59}Ni$ scaling factors are consistently high, with a rather large percentage of the factors outside the range in Table B.1.

These analyses indicate that the SAIC use of a generic $^{59}Ni/^{63}Ni$ scaling factor is likely to overestimate $^{59}Ni$ concentrations.

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This report evaluates the techniques used by industry to characterize and classify irradiated hardware components for disposal in accordance with the requirements of 10CFR Part 61 and the uncertainties associated with application of these techniques. It describes current methods and practices, identifies the uncertainties associated with the methods and practices considered, and recommends areas for improvement which could reduce uncertainty.

Two different characterization methods are described. The first uses a combination of gamma scanning, direct sampling, underwater radiation profiling and radiochemical analysis to determine radionuclide content, while the second uses a form of activation analysis in conjunction with underwater radiation profiling. Both methods employ two distinct steps; (1) the determination of Cobalt-60 content and (2) the determination of scaling factors for hard-to-detect Part 61 radionuclides. Current uncertainties in Cobalt 60 determination can be reduced by improving underwater radiation profiling equipment and techniques. The calculational techniques used for activation analysis can also be refined to reduce the uncertainties with Cobalt-60 determination. Improved radiochemical analysis techniques, and the development of a statistically significant set of component composition data would eliminate some of the conservatism in current scaling factor determination.

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